NUREG-0054
(Supplement No. 1 to NUREG-75/100)

Safety Evaluation Report

U. S. Nuclear
Regulatory Commission

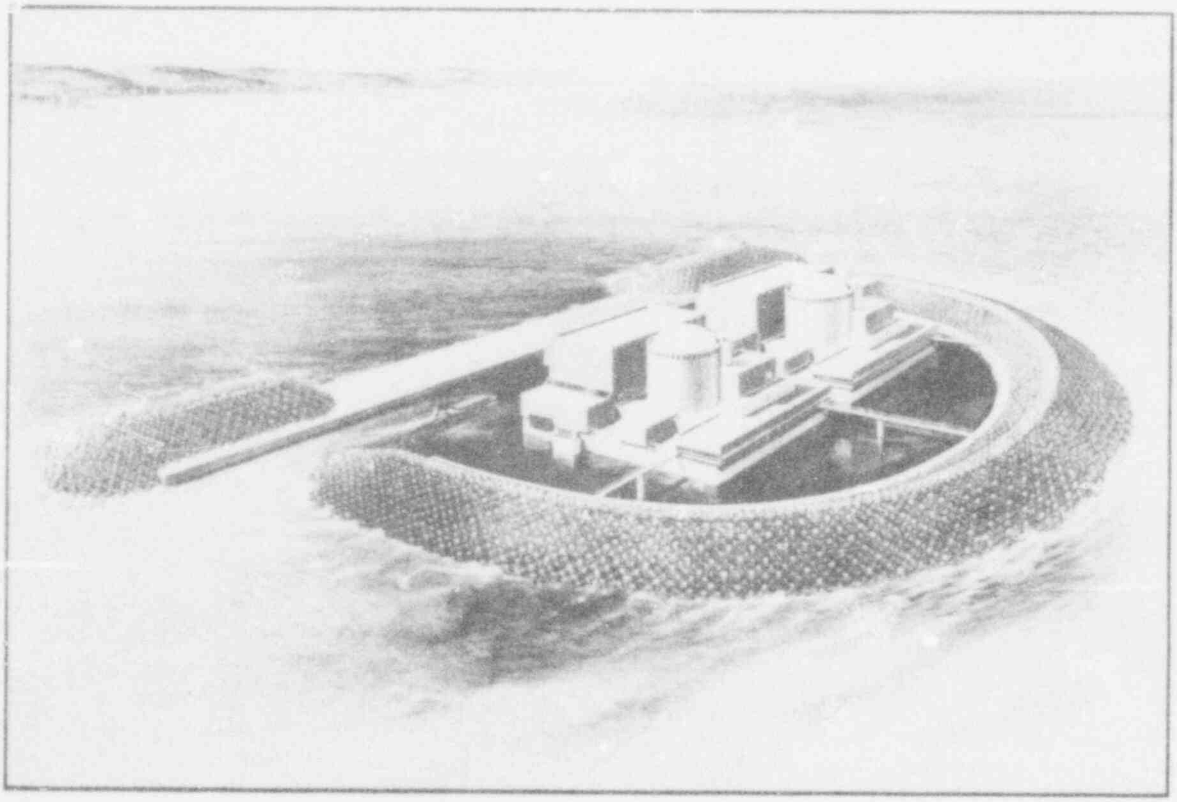
related to operation of

Office of Nuclear
Reactor Regulation

Offshore Power Systems Floating Nuclear Plants (1—8)

Docket No. STN 50-437

March 16, 1976



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NUREG-0054
Supplement No. 1
to NUREG-75/100

SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

OFFSHORE POWER SYSTEMS

FLOATING NUCLEAR PLANTS (1-8)

DOCKET NO. STN 50-437

MARCH 16, 1976

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 General Background

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report in the matter of the application of Offshore Power Systems (hereinafter referred to as the applicant) for a license to manufacture eight standardized floating nuclear plants was issued on September 30, 1975. In that Safety Evaluation Report we identified (1) two matters requiring additional information from the applicant, (2) three matters where our review of information submitted by the applicant was not yet complete and (3) six matters wherein the applicant's proposed design differed from staff requirements.

The purpose of this Supplement is to update the Safety Evaluation Report by providing (1) our evaluation of additional information submitted by the applicant since the Safety Evaluation Report was issued, (2) our evaluation of the matters where we had not completed our review of information submitted by the applicant when the Safety Evaluation Report was issued and (3) our responses to the comments made by the Advisory Committee on Reactor Safeguards in its report dated December 10, 1975.

The areas of primary U.S. Coast Guard review and inspection responsibilities are included in the Safety Evaluation Report and this Supplement. These areas are delineated in the "Memorandum of Understanding Between the U.S. Coast Guard and The U.S. Atomic Energy Commission for Regulation of Floating Nuclear Power Plants," January 4, 1974. The U.S. Coast Guard will issue a letter of acceptance indicating its satisfaction with the preliminary design information relating to its review of the application.

Except for the appendices, each of the following sections of this Supplement is numbered the same as the section of the Safety Evaluation Report that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the Safety Evaluation Report. Appendix A is a continuation of the chronology of the staff's principal actions related to the processing of the application. Appendix B is the Interim Report of the Advisory Committee on Reactor Safeguards on the Floating Nuclear Plant. A communication from the National Oceanic and Atmospheric Administration is included as Appendix C.

1.6 Site-Related Design Envelope

The site-related design envelope parameters are summarized in Table 1.2 of the Safety Evaluation Report. This table has been revised to reflect our present evaluation. For convenience, the revised table has been reproduced in its entirety in this Supplement. Except for items (12), (13) and (14) the table is essentially the same as the table in the Safety Evaluation Report. Additions or changes are identified by a vertical margin bar.

1.10 Outstanding Issues

In Section 1.10 of the Safety Evaluation Report, we listed a number of outstanding issues. All of the outstanding issues have been resolved with the single exception that evaluation of emergency core cooling system design in accordance with Appendix K to 10 CFR Part 50 is not yet complete (Section 6.3.3). The resolution of this matter will be reported in a future supplement to the Safety Evaluation Report.

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TABLE 1.2 (REVISED)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(1) Vital areas must not flood during the postulated sinking emergency	Maximum mean low water depth (Note 1)	Basin water depth at mean low-water must satisfy all of the following conditions (Note 2): (a) Mean low water ≤76 ft *minus* 10 percent exceedance high spring tide *minus* 1/100 year storm surge *minus* allowance for wave crest adjacent to vital structures. (b) Mean low water ≤76 ft *minus* 10 percent exceedance high spring tide *minus* maximum tsunami *minus* allowance for wave crest adjacent to vital structures.	2.3
(2) Plant must not ground under the influence of environmental loads	Minimum mean low water depth (Note 1)	Basin water depth at mean low water must satisfy all of the following conditions (Note 3): (a) Mean low water ≥ Plant Draft *plus* maximum downward displacement produced by the design basis tornado. (b) Mean low water ≥ Plant Draft *plus* 10 percent exceedance low spring tide *plus* drawdown from stillwater level produced by the probable maximum hurricane *plus* maximum downward corner displacement produced by the probable maximum hurricane at conditions of maximum storm drawdown. (c) Mean low water ≥ Plant Draft *minus* 10 percent high spring tide *minus* storm surge produced by the probable maximum hurricane *plus* maximum downward corner displacement produced by the probable maximum hurricane at conditions of storm surge. (d) Mean low water ≥ Plant Draft *plus* 10 percent exceedance low spring tide *plus* drawdown produced by tsunami.	2.3

TABLE 1.2 (REVISED) (Continued)

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(3) Plant design basis motion must not be exceeded	(a) Plant response spectra at four specified locations (expressed in terms of equivalent static accelerations)	(a) Horizontal Component: (1) Probable maximum hurricane, 0.10g (2) Tornado with continuous basis motion, 0.10g (3) Safe shutdown earthquake with continuous basis motion, 0.20g Vertical Component: (1) Probable maximum hurricane, 0.10g (2) Tornado with continuous basis motion, 0.10g (3) Vertical component due to horizontal safe shutdown earthquake with continuous basis motion, 0.05g	3.7.1
	(b) Ground response sprectra	(b) Vertical component only, safe shutdown earthquake, 0.20g	
	(c) Maximum design basis angular displacement about any axis in the horizontal plane due to combined pitch and roll (Note 4)	(c) 3 degrees	
	(d) Ground response spectra with plant in sunken condition	(d) Horizontal Component: operating basis earthquake, 0.15g Vertical Component: operating basis earthquake, 0.10g	
(4) Plant operating bais motion must not be exceeded during operating basis events	(a) Plant response spectra at four specified locations (equivalent static accelerations)	(a) Horizontal Component: (1) Operating basis earthquake with continuous basis motion, 0.10g (2) Operating basis wind and wave, 0.05g	3.7.1

TABLE 1.2 (REVISED) (Continued)

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
		Vertical Component: (1) Vertical component due to horizontal operating basis earthquake with continuous basis motion, 0.025g (2) Operating basis wind and wave, 0.05g	
	(b) Maximum operating basis angular displacement about any axis in the horizontal plane due to combined pitch and roll (Note 4)	(b) 2 degrees	
(5) Plant continuous basis motion must not be exceeded during continuous basis wind and wave	(a) Plant response spectra at four specified locations (expressed in terms of equivalent static accelerations)	(a) Horizontal Component: Continuous basis wind and wave, 0.015g Vertical Component: Continuous basis wind and wave, 0.015g	3.7.1
	(b) Maximum continuous basis angular displacement about any axis in the horizontal plane due to combined pitch and roll	(b) 0.5 degrees	
(6) Pressure loads on the plant superstructures must not exceed the design value	(a) Tornado	(a) Rotational speed: 290 miles per hour Translational speed: 70 miles per hour (maximum), 5 miles per hour (minimum); Pressure drop: 3.0 pounds per square inch.	3.3 & 3.8
	(b) Design basis wind (probable maximum hurricane)	(b) Fastest mile wind speed, 204 miles per hour	
	(c) Operating basis wind	(c) Fastest mile wind speed, 160 miles per hour	
(7) Basin water must not experience a "hard freeze"	Basin Ice	Continuous sheet of basin ice must not occur or must be prevented by utility-owner action.	2.7.3
(8) Maximum basin water temperature must not exceed the design basis of safety-related cooling water system.	Maximum basin water temperature	85 degrees Fahrenheit	2.7.3

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TABLE 1.2 (REVISED) (Continued)

	Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(9)	Minimum air temperature at the sea surface (0-5 meters) must not be less than the design service temperature of the hull steel	Air temperature	-5 degrees Fahrenheit	2.7.2
(10)	Minimum basin water temperature must not be less than the design service temperature of the hull steel	Minimum basin water temperature	28.6 degrees Fahrenheit	2.7.3
(11)	Precipation must not overload plant roof structures	Precipitation rate (rainfall or waterspout)	13 inches per hour	2.7.6
(12)	A class of accidents, the consequences of which could exceed the plant design basis, must have a low probability of occurrence	a) P (aircraft crash) b) P (flammable vapor cloud) c) P (toxic chemical spill) d) P (explosion >2 pounds per square foot reflected overpressure) e) P (toxic vapor cloud)	(a) through (c): $P \leq 10^{-7}$/year (d) $P \leq 10^{-7}$/year, or demonstrate site features prevent explosion from occurring near enough to the plant to produce >2 pounds per square foot reflected overpressure (e) $P \leq 10^{-7}$/year or demonstrate that concentration of toxic vapor at control room and emergency relocation area intakes does not exceed limits given in Table 2.9.1	2.9
(13)	Accident dose offsite must not exceed 10 CFR 100	While body dose; thyroid dose	The combination of plant accident releases, atmospheric diffusion, exclusion boundary radius, and low population zone radius must result in doses less than or equal to 10 CFR 100 limits. For determining exclusion boundary, the two-hour χ/Q value at the boundary should be 1.9×10^{-3} sec/m^3 or less	2.8.2
(14)	Normal operating doses must not exceed 10 CFR 50, Appendix I	Whole body dose and thyroid dose from gaseous effluents; dose from liquid effluents	The combination of normal plant operating releases, atmospheric diffusion, and site boundary must result in doses less than or equal to 10 CFR 50, Appendix I limits for gaseous effluents; doses from liquid effluents must be less than or equal to 10 CFR 50, Appendix I limit.	2.8.1

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TABLE 1.2 (REVISED) (Continued)

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(15) Basin floor must be adequate to support the plant in the sunken condition	(a) Flatness deviations	(a) $\leq$2 foot from mean plane and $\leq$10 foot in-plane extent	2.5.2.1
	(b) Bearing strength	(b) 1600 pounds per square foot	
(16) The mooring system must: (a) Transmit loads at the plant mooring foundations	(a) location of plant/mooring system	(a) Five feet above plant bottom near the corners of the plant	2.6
(b) not overload the plant mooring foundations	(b) transmitted mooring system loads	(b) To be specified during detailed design	
(c) allow level and non-level sinking	(c) mooring system	(c) 0 to 6 degrees sinking	
(17) Plant must be prevented from colliding with site structures	Site configuration, mooring system and other site structures	Site dependent	2.6 and 2.10.2
(18) A reliable source of offsite power must be provided	(a) Separation and availability of circuits	(a) General Design Criterion 17	2.10.1
	(b) Number of circuits	(b) General Design Criterion 17 or as required for continuity of alternating current power, whichever is greater	
	(c) Integrity of the power connection with the plant	(c) Must remain functional during operating basis events experienced at the specific site	
(19) Either the onsite or offsite alternating current power system must be continuously available	The combined probability of (1) a loss of offsite power for a period in excess of seven days and (2) inability to replenish diesel fuel during a continuous seven-day period conincident with the loss of offsite power	P $\leq 1 \times 10^{-7}$ per year	2.10.1
(20) A fuel oil spill occurring outside the site structure must be prevented from reaching a point closer than 100 feet from the plant	Site protective structure	100 feet from plant	2.9.4.1

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TABLE 1.2 (REVISED) (Continued)

	Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(21)	Site design basis accidents and environmental conditions must not produce missiles which prevent achieving safe shutdown	Site missiles	Impact or penetration equal to or less than: (a) 25 ton boat, 50,000 pounds (b) Wood plank, 4 inches x 12 inches x 12 feet, 200 pounds (c) Steel pipe, 3 inches diameter, schedule 40, 10 feet long, 78 pounds (d) Steel rod, 1 inch diameter, 3 feet long, 8 pounds (e) Utility pole, 13-1/2 inches diameter, 35 feet long, 1,490 pounds	2.9 and Table 3.5.1
(22)	Vessels which can penetrate the first inboard bulkhead or breach more than two watertight compartments must be prevented from striking the plant with a velocity that would cause this damage	Site structure	Impact on the plant equivalent to a ship of 3,500 tons at 13 knots	2A.8
(23)	Operating basis wave in the basin must not exceed the operating basis value for the platform hull	Waves in basin	The mean wave height between crest and trough associated with a wave length between 350 and 550 feet must not exceed 6 feet	3.12.2.2.1
(24)	Design basis wave in the basin must not exceed the design basis value for the platform hull	Waves in basin	The mean wave height between crest and trough associated with a wave length between 350 and 550 feet must not exceed 10 feet	3.12.2.2.1
(25)	Corrosion of the immersed surfaces of the platform hull must be controlled by a suitable cathodic protection system	(a) Minimum post-polarization, current-off negative hull	(a) -0.85 volts (Versus copper-copper sulfate reference electrode)	9.6.3
		(b) Polarization capacity	(b) Achieve polarization within 60 days at 90 percent current capacity taking into account stray currents Maintain polarization at 75 percent current capacity taking into account stray circuits	
		(c) Redundancy/reserve capacity	(c) Maintain polarization with single component failure taking into account stray currents	

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TABLE 1.2 (REVISED) (Continued)

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
	(d) Number of rectifiers/anode groups	(d) 8 minimum	
	(e) Rectifier control	(e) Automatic by hull-mounted reference electrodes	
	(f) Interference from other structures	(f) Eliminate by bonding together electrically all submerged steel structures	
	(g) Performance monitoring	(g) Program to be implemented by owner	

Note (1): The equations in the "Envelope Parameter Limit" column define limits of acceptable mean low water (MLW) depth which must be satisfied throughout the life of the plant. Deviations from the nominal elevation of the basin floor at each specific site must be taken into account in order to determine the range of water depths at MLW which might be encountered during the life of the plant; expected maximum and minimum MLW depths are then compared to the limits established by the above equations.

Note (2): For river sites, the site characteristics that need to be combined and compared to the 76 feet maximum water depth are:

Operating Basis Flood level in basin (Standard Project Flood)
+Operating Basis Storm Surge in basin (1 in 100 year storm)
+Allowance for wave adjacent to vital structures

Note (3): Including static trim in addition to motion produced by environmental loading.

Note (4): It is not an implied requirement that the minimum MLW depth at all sites accommodate the platform corner displacement associated with 3 degrees.

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2.0 PLANT-SITE INTERFACES

2.6 Mooring System

We stated in the Safety Evaluation Report that is is technically feasible to design and build a satisfactory mooring system and that we would evaluate the adequacy of each mooring system as proposed by the utility-owner of the plant. A total of twelve mooring system anchor points will be provided on three sides of the hull structure. The specific anchor point load components are specified in the Plant Design Report. The mooring system anchor points are discussed in Section 3.11.5 of the Safety Evaluation Report. Each utility-owner may utilize as many of the anchor points specified on the hull as required by its particular mooring scheme. Each plant may be moored differently and may include different degrees of redundancy depending upon the margins of safety used in the design. We will evaluate the adequacy of each mooring system, including its degree of redundacy, as proposed by the utility-owner of each plant.

2.8 Site Environment

2.8.1 Meteorology

2.8.1.1 Regional Climatology

In the Safety Evaluation Report we stated that our design requirement for the operating basis sustained wind speed at a height of 30 feet above sea level with a return period of 100 years is 160 miles per hour. This requirement is based on data provided by the National Oceanic and Atmospheric Administration (see Appendix C).

The applicant in Amendment 21 to the Plant Design Report has stated that the plant will be designed for an operating basis wind speed of 160 miles per hour. We consider this matter resolved.

2.8.2 Wind Convergence Over A Breakwater

In the Safety Evaluation Report it was noted that the applicant proposed a series of wind tunnel tests to determine the effects of convergence over the breakwater on wind loads on the plant. Also, the applicant committed to designing the plants to accommodate any increase in loadings indicated by these tests. The results of the tests are presented in the applicant's Report No. TR-16159, "Wind Tunnel Study of Wind Forces on a Floating Nuclear Power Plant," which was reviewed by the staff and found to be acceptable.

The tests included four breakwater configurations, (1) a one on two slope 66 feet high, (2) a one on four slope 66 feet high, (3) a one on two slope 90 feet high and (4) a vertical wall 66 feet high and 24 feet wide. The results of the wind tunnel tests showed a reduction in loads on the structures due to the presence of any of the four breakwater configurations tested in comparison to the case without a breakwater and therefore no change in plant design will be required. However, for breakwater configurations outside the range of those tested, the utility-owner will be required to evaluate the effects of convergence over the breakwater.

2.10 Site Accidents

2.10.2 Shipping Accidents

(b) Oil/Gasoline Tanker Collision

In response to our concern regarding explosions associated with petroleum tanker accidents, the applicant submitted additional material reporting on a study of the hazard to the floating nuclear plant from petroleum tanker accidents which included empty (i.e., vapor filled) tankers, initially full tankers, vapor clouds from heated tanks, and vapor clouds from floating spills. In addition, the study examined the probability of occurrence of these various accidents for a representative site.

The explosion of a vapor filled tanker was identified as the petroleum tanker accident with potentially the most severe consequences. Using conservative assumptions, the applicant calculated the size of the petroleum tanker explosion which would create an overpressure equal to the plant design criteria (2 pounds per square inch reflected overpressure) as a function of distance from the floating nuclear plant. For example, the explosion of a 10,000 dead weight ton tank barge in ballast at a distance of about 540 feet or a 25,000 dead weight ton coastal tanker in ballast as a distance of about 740 feet would meet the plant design overpressure criteria. We are in agreement with the overpressures calculated by the applicant for petroleum tanker explosions. The applicant's study also indicated that the probability of this type of accident occurring in the near vicinity of a floating nuclear plant at a representative site was extremely low.

The site design envelope parameter referring to explosive accidents external to the plant states that for a site to be acceptable it must be demonstrated that the probability of an explosion which produces a reflected overpressure of 2 pounds per square inch or greater on the plant's Category I structures is of the order of 10^{-7} per year or less. We will require that each utility-owner applicant demonstrate that the selected site possesses protective features which provide adequate separation distance to insure that the plant design blast overpressure criteria will not be exceeded or that the probability of such an event, based on a detailed study of the local shipping traffic and other local hazard sources is of the order of 10^{-7} per year or less.

Our conclusion is that floating nuclear plant sites which meet these stipulations are not unduly threatened by petroleum tanker explosions.

With regard to missiles, we have analyzed the maximum range expected of missiles generated by explosions within tankers and have concluded that the range of potentially damaging missiles from petroleum tanker explosions is less than that of the potentially damaging blast overpressures.

For service vessels used to supply fuel oil to the floating nuclear plant, the utility-owner has the responsibility of insuring that an accident during fuel supply operations does not produce overpressures which exceed the plant design criteria.

We will require that the utility-owner demonstrate that explosions of unacceptable magnitude are prevented from occurring during fuel supply operations by such means as limiting the capacity of the individual tanks on the supply barge or by providing safeguards such as tank inerting or adequate separation distance between the fuel off-loading facilities and the plant.

3.0 DESIGN CRITERIA FOR STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.5 Missile Protection Criteria

3.5.1 Tornado Missiles

In the Safety Evaluation Report we stated that we require that the design of the floating nuclear plant meet our tornado missile criteria. The applicant in Amendment 21 to the Plant Design Report has indicated that the plant will be designed to meet our tornado missile criteria. We consider this matter resolved.

3.8 Design of Seismic Category I Structures

3.8.1 Containment (Steel)

In the Safety Evaluation Report we stated that the summary of the buckling criteria was acceptable. We also indicated that we would review the details and implementation of the criteria when provided by the applicant. The applicant has submitted Report No. 7270-RP-16A51, "Buckling Criteria and Application of Criteria to Preliminary Design of the Steel Containment Shell for the Floating Nuclear Plant," dated December 15, 1975. We have discussed our review of the report and the methodology utilized in the application of the buckling criteria with the applicant. As a result of our review we find the report to be acceptable, however, the applicant has agreed to revise the report to provide elaboration and clarification on detailed implementation matters. In addition, in its letter of March 2, 1976, the applicant has agreed to provide an independent confirmation of its buckling analysis. We consider this matter resolved.

3.8.4 Air Blast Procedures

The applicant has stated that the Category I structures protecting equipment required for safe shutdown will be designed to withstand a reflected overpressure of 2 pounds per square inch. We have reviewed the applicant's criteria which will be utilized to assure that the plant structures are capable of withstanding these forces. The applicant has agreed to the staff requirement that air blast loads must be determined by elastic dynamic analysis. Air blast loads are combined with other concurrent design loads in the same manner as tornado loads. Allowable limits are determined from either the Standard Review Plan Section 3.8.4 or the ASME Boiler and Pressure Vessel Code, Section III, Appendix F(F-1323).

Use of these procedures provides assurance that adequate margins of safety will be maintained to resist the effects of accidental explosions near the floating nuclear plant. We conclude that the procedures delineated in Appendix 3G of the Plant Design Report are acceptable.

3.11 Platform Structure

We have reviewed the design of the platform as it relates to verification of the structural design of the plant. The platform is a highly redundant structure consisting of thousands of elements and numerous watertight compartments. Extensive experience exists in the design of floating structures such as drill rigs, ocean-going barges, supertankers, submarines and aircraft carriers, which can be relied upon to assure the high degree of structural reliability required of the platform for the floating nuclear plant. The applicant is performing a detailed three dimensional finite element analysis consisting of thousands of elements in order to assure the adequacy of the design. The applicant also intends to take deflection and draft measurements during the construction of the plant and compare these measurements with the values predicted by their calculations. In addition to the final safety review which will be conducted by the Nuclear Regulatory Commission, the U.S. Coast Guard will further assure the adequacy of the design by reviewing the detailed design drawings and calculations by performing their own independent analysis of the platform structure. The detailed analytical programs, the redundancy of the platform structure, the deformation and draft measurements during construction and the vast experience in the design and construction of floating structures will provide adequate assurance that the platform will be an

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extremely reliable structure. On the basis of the above, we have not felt that it is warranted to require further verification of the design by means of platform strain and deformation measurements.

The plant will be instrumented to monitor plant motions during tow to provide a means of determining whether plant systems and components are overstressed. Instrumentation will be provided to supply data on the motion of the plant due to wind, waves and earthquakes during operation of the plant. In its review of the Atlantic Generating Station application, the staff has also discussed provisions for monitoring the forces in the mooring system and the ability to correlate these forces with the plant motions. We intend to evaluate the need for such instrumentation programs with the utility-owner of each plant.

The plans for visual inspection and nondestructive testing of the platform structure are described in Chapter 3 of the Plant Design Report and in particular Section 3.12.6 (Corrosion Control) and 3.12.7 (Inspection and Maintenance After Construction). They are further amplified in the applicant's Report No. AD-7100-14A85, "FNP Platform Hull Dry Docking Equivalency." The staff and the U.S. Coast Guard evaluation of these plans is discussed in Section 3.11 of the Safety Evaluation Report.

We conclude that the design criteria and design controls discussed above and presented in the Plant Design Report provide adequate verification of the structural design. The staff will further verify the adequacy of the design in its review of the applicant's Final Design Report and will determine then if there need be any additional requirements.

3.11.1 Hull Material

We stated in the Safety Evaluation Report that we require that the design criteria for the hull material include Charpy-V-Notch procedure qualification testing. The applicant in Amendment 21 to the Plant Design Report revised the design criteria for the hull material to meet our requirements. The applicant will also undertake a test program to establish the suitability of the Dynamic Tear test in the heat affected zone. The test program and results will be submitted to the Nuclear Regulatory Commission and the U.S. Coast Guard for review and approval. If the test program and results prove conclusively that the Dynamic Tear test can be used in the heat affected zone, the applicant proposes to substitute it for the Charpy-V-Notch testing for qualification and production testing. We find this acceptable. We consider this matter resolved.

3.11.3 Corrosion Control

We have examined the platform hull splash zone corrosion protection system with regard to the practicality of repair or renewal. The splash zone has severe protection requirements because continual wetting of the surface by aerated sea water is alternated with exposure to the atmosphere. It is recognized that the hull coating will not have unlimited life and that maintenance will be necessary. This has been anticipated and provision has been made for this eventuality.

The applicant proposes a silica-filled catalized epoxy coating that has a high tolerance of wet conditions during coating application. In addition, the nature of the coating is such that local repairs can be made underwater. However, if extensive repairs are necessary or if recoating is indicated, this can be facilitated by trimming the platform. The platform trim system is designed to provide controlled ballasting of $\pm$ 1 degree for maintenance condition. Alternately, cofferdam techniques could be employed without the need for tilting the platform.

We therefore conclude that a splash zone corrosion control system that may have a life of less than 40 years may be used, since adequate means are available to perform repairs or recoating without causing deviation from the floating nuclear plant design limits.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In Amendment 19 to the Plant Design Report the applicant increased the maximum temperature of the ultimate heat sink from 85 degrees Fahrenheit to 95 degrees Fahrenheit to allow expanded/riverine sitings of a floating nuclear plant. This change required the design capacity of the containment spray pumps to be increased from 2400 gallons per minute to 2800 gallons per minute. The containment spray heat exchangers and system piping have also been upgraded, and the flow area of the containment sump screen assemblies has been increased. All containment analyses sensitive to the changes in the containment spray system design have been repeated. The results indicate that:

(1) the margin between the available net positive suction head at the spray pump inlet and the required net positive suction head has increased from 72 percent to 80 percent of the required net positive suction heat;

(2) the containment capability with regard to bypass steam flow from the containment lower compartment to the upper compartment is essentially unchanged; and,

(3) the maximum calculated containment pressure has decreased from about 12.5 pounds per square inch, gauge to about 12.2 pounds per square inch, gauge.

Our previous conclusion that the designs of the primary containment and the containment heat removal systems are in accordance with the appropriate General Design Criteria remains unchanged.

6.2.4 Containment Isolation Systems

In the Safety Evaluation Report for the floating nuclear plant, we reported that the applicant proposed the use of simple check valves outside containment as isolation valves for the main and auxiliary feedwater lines. The use of simple check valves outside containment for this purpose is expressly prohibited by General Design Criterion 57.

In discussions with the applicant regarding this design feature, the applicant committed to upgrade the design of the main feedwater line to seismic Category I and ANS Safety Class 2 from the containment penetration up to and including the feedwater regulator valve, and the design of the auxiliary feedwater line to seismic Category I and ANS Safety Class 2 from the containment up to and including the auxiliary feedwater stop valves. The applicant has further committed to reclassify the feedwater regulator valve and the auxiliary feedwater stop valves as containment isolation valves. These commitments were reported in the November 7, 1975, meeting of the ACRS and have been documented in Amendment 21 to the Plant Design Report for the floating nuclear plant.

We therefore conclude that containment isolation provisions for the main and auxiliary feedwater lines are in compliance with the requirements of General Design Criterion 57 and are acceptable.

6.2.8 Main Steam Line Break Inside Containment

We have reevaluated the floating nuclear plant with regard to the containment pressure and temperature response to a main steam line break inside containment. Our recent review of the Westinghouse Electric Corporation LOTIC-1 and LOTIC-2 codes revealed the method of calculating heat transfer from a superheated steam environment to passive heat sinks in the containment lower compartment to be not conservative. The LOTIC-1 code was used by the applicant to analyze the containment pressure and temperature response to a main steam line break. In a recent communication the applicant indicated recognition that the LOTIC-1 code is not capable of accurately calculating the containment temperature and pressure in the superheated steam region. The Westinghouse Electric Corporation is currently modifying the LOTIC-2 code to correct the heat transfer calculations for the lower compartment volume. In its letter of

February 27, 1976, the applicant has committed to reanalyze the steam line break accident and determine the resulting containment temperatures using an appropriate code which we have found to be acceptable.

With regard to containment pressure resulting from a postulated steam line break inside containment, it is our judgment that the containment design pressure of 15 pounds per square inch, gauge will not be exceeded. The maximum containment pressure calculated for the containment prior to complete meltout of the ice condenser is 9.9 pounds per square inch for the design basis loss-of-coolant accident. The containment peak pressure prior to complete ice melt is a function of the mass and energy release rates into the containment. Since the loss-of-coolant accident mass and energy release rates are more than double the mass and energy release rates for a steam line break, the containment pressure will not exceed 9.9 pounds per square inch as long as there is ice in the ice condenser. The applicant has provided sufficient information to show that, considering single failure in the feedwater system and manual isolation in the auxiliary feedwater system, flow from the steam line will be terminated prior to complete ice melt in the ice condenser, and as a result, the containment design pressure would not be violated for a main steam line break inside containment.

We therefore conclude that the applicant's commitment to reanalyze the containment response when an acceptable code is available is acceptable at this stage (preliminary design) of the licensing process.

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

The evaluation of emergency core cooling system design in accordance with Appendix K to 10 CFR Part 50 is not complete. Our evaluation and conclusions will be included in a supplement to the Safety Evaluation Report.

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7.0 INSTRUMENTATION AND CONTROL

7.3 Engineered Safety Features Actuation System

7.3.5 Application of the Single Failure Criterion To Manually-Controlled, Electrically-Operated Valves

We are currently performing a generic review of a Westinghouse proposal to eliminate from the design basis those single failures in emergency core cooling system valve control circuitry that result in spurious valve actuation. The resolution of this issue is directly applicable to the hot leg injection valves in the floating nuclear plant design. In Amendment 21 the applicant committed to conform to the generic Westinghouse Electric Corporation resolution of this matter. On the basis of the applicant's commitment, we conclude that the design is acceptable.

7.5 Safety-Related Display Instrumentation

We stated in the Safety Evaluation Report that the design criteria for the safety-related display instrumentation are presented in RESAR-3, Section 7.5, including instrumentation for post-accident monitoring and safe shutdown. Regarding this aspect we consider that the range of safety-related display information is adequate to enable the plant operator to take correct action during and after an accident. The indicators and recorders referenced in these secions will be mounted in the main control room in a manner consistent with the functional requirements of plant operation. All information and control facilities required during the course of an accident and post accident recovery will be located in an area within the control room that will be utilized exclusively for accident mode operation. In those cases when the information displayed for accident monitoring is also required for normal operation, the same information channel will be employed. The information displays required for normal operation will be identical in range and format to those used for accident monitoring and will be located in the "normal operation area" of the control room. We therefore conclude that the proposed design of the safety-related display instrumentation meets our requirements and is acceptable.

8.0 ELECTRIC POWER SYSTEMS

8.2 Offsite Power System

The design of the offsite power system includes two physically independent 100 percent capacity auxiliary transformers to be used with two physically independent plant-to-shore transmission circuits arranged in such a manner as to minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions and so that both are immediately available in the event of loss of one offsite circuit. In addition, the design provides for manual switching capability to connect a third transmission circuit (cable) in place of either of the normal operating circuits to be used in the event of extended loss of one offsite cable. The design of the transmission circuits, including the flexible connection, however, is outside the scope of the Plant Design Report and will be evaluated pursuant to a specific utility-owner site.

Therefore we conclude that the design of the offsite power terminations and associated circuitry provided in the design of the floating nuclear plant satisfies General Design Criteria 17 and is acceptable.

8.4 Physical Independence of Electric Systems

Additional information has been provided by the applicant regarding fire prevention and control. The applicant has indicated that the cable design for the floating nuclear plant was selected to provide an optimum balance of electrical, physical, aging, and water absorption characteristics, and flame retardant and mechanical properties. The cable flame retardancy is considered capable of providing acceptable margins in excess of its postulated fire exposure. Power and control cable insulation will be ethylene-propylene rubber-base with an overall jacket of neoprene or hypalon or will be an insulation having properties equal to or better than the above insulation. Instrumentation cables will vary with the type of signal conveyed but will meet the insulation properties of power and control cables. Power and control cables have been subjected to Underwriter's Laboratories (UL), National Electrical Manufacturers Association, Institute for Power and Control Engineers Association, and other flame tests to prove cable reliability.

Fire retardant wiring will be utilized throughout the control boards for both redundant and non-redundant circuits as an additional safety factor.

Cable penetrations through fire rated walls and floors will be designed and constructed such as to maintain the barrier integrity without transmitting flame for the rating duration. Design criteria which are presently being developed by industry (for example, the Institute of Electrical & Electronic Engineers Power Generation Committee) will be reviewed and evaluated for adoption as appropriate to all stop points. Design consideration will address but not be limited to:

(1) Stop material and its rating characteristics.
(2) Test methods and qualifying procedures.
(3) Installation quality assurance procedures.
(4) Modification procedures (adding cables after stop installation).
(5) Suggested periodic inspection procedure.

The floating nuclear plant design includes a fire protection system designed to prevent, detect, extinguish, limit or control fire and its hazards and damaging effects, both inside the floating nuclear plant and inside a breakwater basin (also see Section 9.5.1 of this Supplement). All areas within the floating nuclear plant which contain hazardous materials, vital equipment, or equipment important to safety will be protected from fire exposure by either, or a combination of, fire resistive barriers, spatial separation, or fire detection and automatic and manual extinguishing systems. Automatic wet pipe sprinkler systems will be provided in areas of high cable density such as the

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cable pull area and the containment electrical penetration areas. Manually actuated carbon dioxide fire protection systems will be provided for the rooms housing the safety related control and instrumentation racks and the diesel generators.

A fire and smoke detection alarm system provided in the design will give immediate audible and visual alarms. This system will also monitor the status of the automatic fire extinguishing systems The design philosophy used seeks rapid identification of the location of a fire so that corrective measures may be taken to limit damage. The monitored regions of the plant are divided into functional areas. The detection system for each area will be independent of every other area, except for a common alarm panel in the control room. Our review of the design of electrical control and instrumentation systems important to safety included consideration of potential fire propagation to redundant safety systems. We conclude that the proposed design criteria and commitments in this regard meet present staff requirements and are acceptable.

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9.0 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

Subsequent to the issuance of the Safety Evaluation Report the applicant provided additional information regarding its fire protection and detection system design. Our review of this information is summarized in the following paragraphs.

The applicant has initiated a test program to verify suitability of the external wall and weir material to withstand external fires, resulting from a service barge accident within 100 feet of the plant, and from a petroleum tanker spill and ensuing fire occurring beyond 100 feet from the plant.

The system design to be tested consists of a weir designed to distribute cooling water on the exterior metal wall surfaces. The test result will be used to establish curves of heat flux versus time, and wall temperature versus time for a spectrum of fires using various materials of construction. The applicant has stated that the staff and U. S. Coast Guard would be kept informed of the progress of the tests program and of design developments.

An external fire detection system to alarm in the control room has been incorporated in the design of the plant. The applicant has committed to a test program to determine heat rate and wall temperature curves for sustained heat fluxes which will be used to establish the location and type of detection equipment necessary to protect the plant from such external exposure fires.

The design of the internal fire detection and alarm systems is based on the use of monitored zones. The detectors will be primarily located in unmanned areas not protected by automatic fire extinguishing systems. Standpipe hose stations will be located at all elevations so that all parts of the plant are within reach of two hose streams from different hydrants. The applicant has stated that the final design of the fire protection and detection system will reflect considerations of the recommendations of the staff report, "Recommendations Related to Browns Ferry Fire," NUREG-0050, February, 1976.

Based on our review of the systems for detecting and protecting against fires, both internal and external to the plant, and conformance to the requirements of General Design Criterion 3, we conclude that the design criteria and proposed test programs are acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

10.2.1 Design Considerations for Floating Nuclear Plant

The platform hull does not provide the rigid base for the turbine foundation found in a conventionally constructed power plant. As a result, the applicant has evaluated the turbine generator design in light of the extremes in deflection that it will experience due to platform hogging and sagging. The analysis also included the inertial and gyroscopic forces associated with the platform motions. Although not completed, the preliminary results have indicated the ability of the turbine generator/turbine foundation system to accommodate the anticipated platform deflections and motions. To minimize the effects of platform deflections transmitted to the turbine-generator machinery, selective alignment of the turbine-generator will be done. This procedure is similar to that used on land based plants, where the effects of operating conditions such as vacuum loads, are compensated for in alignment of the machinery during erection. In addition, as stated in the Safety Evaluation Report, the applicant has committed to analyze and test the turbine generator/turbine foundation/platform system to verify that the turbine foundation adequately decouples the turbine generator from the platform so as to minimize the turbine-induced vibrations in other components.

We conclude that it is feasible to design and install a turbine generator/turbine foundation system which will function properly on the platform. We will evaluate the final design and verification analysis during our review of the final plant design report.

11.0 RADIOACTIVE WASTE MANAGEMENT

We stated in the Safety Evaluation Report that our evaluation of the capability of the liquid and gaseous radioactive waste treatment systems to meet the dose design objectives of Appendix I to 10 CFR Part 50 was not complete. The applicant, in Amendment 21, stated that the design objective of the plant is to meet the guidelines for quantities of radioactivity released set forth in the Annex to Appendix I (September 4, 1975). Additionally, the applicant in its letter of March 4, 1976, indicated that for the broad siting spectrum, the annual average doses from liquid and airborne activity would also meet the dose guidelines specified in the Annex. The annual average dose estimates for liquid discharges and for discharged airborne activity were evaluated using conservative meteorology. Based on our evaluation of the design capability and design objectives of the radioactive waste management system we conclude that these systems will meet the dose objectives of Appendix I to 10 CFR Part 50 for a broad siting spectrum. Each utility-owner however, will be required to verify and demonstrate that a specific site is in conformance with the plant site design envelope parameter limit.

12.0 RADIATION PROTECTION

During the radiation protection review of the floating nuclear plant application, careful attention was given to the evaluation of whether occupational radiation exposures would be as low as reasonably achievable during operation of the plant. The applicant's preliminary dose assessment gave an estimate of exposure that we found acceptable, when considered at the present stage of the preliminary design. The additional information provided by the applicant in response to comments made by the Advisory Committee on Reactor Safeguards in its December 10, 1975 report indicates that unlimited access areas will be designed to have exposure levels below 0.1 millirem per hour.

In view of the applicant's acceptable proposed implementation of as low as reasonably achievable design criteria, and the additional indication that unlimited access areas will be designed for especially low dose rates, we believe that we can continue to expect that occupational radiation exposures will be as low as reasonably achievable. We will review the calculations and design estimates of specific area dose rates during our review of the final plant design report.

15.0 ACCIDENT ANALYSES

15.4 Radiological Consequences

15.4.1 Loss-of-Coolant Accident Dose Model

In Amendments 19, 20 and 21, the applicant submitted additional information on the secondary containment volumes treated by the annulus filtration system and the exhaust and recirculation flow rates of the annulus filtration system following the occurrence of a postulated loss-of-coolant accident. We have incorporated this information into our loss-of-coolant accident dose model in order to evaluate the radiological consequences of the accident. In addition, the absorption of the low energy beta radiation in the surface tissues of the body was not included in the calculation of the whole body doses. The assumptions and parameters used in our analysis are listed in Table 15.2 (Revised).

Since the floating nuclear plant is a standard plant with no specific site boundary distances or meteorology, we determined the limiting atmospheric dispersion factor (X/Q value) required by a site in order to meet the guideline doses of Regulatory Guide 1.4 - "Assumption Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor". We calculated that a site with a two-hour atmospheric dispersion value of about 1.9×10^{-3} seconds per cubic meter or less at the exclusion area boundary is required to meet both the thyroid and whole body guideline doses of 150 rem and 20 rem, respectively, for the design basis loss-of-coolant accident.* The limiting long term (30-day) atmospheric dispersion values required to meet the guideline doses at the low population zone distance were not determined in this analysis and the suitability of a site with regard to the low population zone doses will be evaluated for each individual floating nuclear plant site. However, based on previous analysis of nuclear power plants of similar size and with similar engineered safety features, and the analysis of the two-hour doses for the floating nuclear plant, there is reasonable assurance that the floating nuclear plant will meet the guideline doses of Regulatory Guide 1.4 at low-population zone distances comparable to those of recently approved sites.

15.4.2 Fuel Handling Accident Dose Model

The radiological consequences of a fuel handling accident have been evaluated based on the limiting two-hour atmospheric dispersion value of 1.9×10^{-3} seconds per cubic meter determined for the loss-of-coolant accident. The assumptions used in the analysis of the fuel handling accident are given, for convenience, in Table 15.3 of this Supplement (identical to that appearing in the Safety Evaluation Report) and the calculated doses (27 rem to the thyroid and 2 rem to the whole body) are shown in Table 15.4 (Revised) of this Supplement.

*Of the sites we have previously evaluated, all of which were on land approximately 90 percent had two-hour dispersion values equal to or less than 1.9×10^{-3} seconds per cubic meter at their exclusion area boundaries.

TABLE 15.2 (Revised)

ASSUMPTIONS USED IN THE CALCULATION
OF LOSS-OF-COOLANT ACCIDENT DOSES

Power Level, megawatts thermal	3579
Operating Time, years	3.0
Containment Volumes, cubic feet	
Lower Compartment	4.08×10^5
Ice Condenser	1.25×10^5
Upper Compartment	7.37×10^5
Primary Containment Leak Rate, percent per day	
0-24 hours	0.5
24 hours	0.25
Bypass Leakage Fraction	2 percent of primary containment leakage
Ice Condenser Elemental Iodine Removal Efficiency	30 percent per pass
Period of effectiveness	0.17 Hour to 0.47 Hour
Containment Spray System	
Effective Volume, cubic feet	7.37×10^5
Removal Rates, inverse hours	
Elemental Iodine	4.5
Particulate Iodine	4.0
Organic Iodine	0
Secondary Containment Volume Treated	
By Annulus Filtration System, cubic feet	6.28×10^5
Mixing Fraction, percent	50
Filter Efficiencies, percent	
Elemental Iodine	95
Organic Iodine	95
Particulate Iodine	99

Annulus Filtration System Flow Distribution

Time Step	Exhaust Flow Cubic feet per minute	Recirculation Flow, Cubic feet per minute
0-10 seconds	0	0
10-300 seconds	6000	2000
300-600 seconds	4500	3500
600-1100 seconds	2600	5400
1100-1700 seconds	500	7500
1700-2700 seconds	2000	6000
2700 seconds - 2 hours	1300	6700
2 - 2.8 hours	350	7650
2.8 hours	250	7750

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TABLE 15.3

ASSUMPTIONS USED IN THE CALCULATION OF FUEL HANDLING ACCIDENT DOSES

Power, megawatts thermal	3579
Number of Fuel Rods Damaged	264
Total Number of Fuel Rods in Core	50,952
Peaking Factor of Damaged Rods	1.65
Shutdown Time, hours	100
Inventory Released from Damaged Rods, percent (Noble Gases and Iodines)	10
Fuel Pool Decontamination Factor	
Iodine	100
Noble Gases	1
Filter Efficiency for Iodines, percent	95

TABLE 15.4 (Revised)

RADIOLOGICAL ACCIDENT CONSEQUENCES

Accident	Exclusion Area Limiting X/Q Value*	0-2 Hour Dose (Rem) Thyroid	Whole Body
Loss-of-Coolant	1.9×10^{-3} seconds per cubic meter	150	20
Fuel Handling	1.9×10^{-3} seconds per cubic meter	27	2

*Required by a site in order to meet Regulatory Guide 1.4 guideline doses (150 rem thyroid and 20 rem whole body) for loss-of-coolant accident.

16.0 MANUFACTURING CONDITIONS

In the Safety Evaluation Report we stated that certain limitations or conditions are required during the manufacture, outfitting and testing of the floating nuclear plant at the manufacturing site to assure integrity and acceptable performance of safety-related features subsequent to nuclear operation and for the service life of the plant. To meet these requirements the applicant has proposed in its letter of March 16, 1976, manufacturing conditions related to manufacture, outfitting and testing of the floating nuclear plant. These conditions include those aspects discussed in Section 16.0 of the Safety Evaluation Report. We have reviewed these conditions and limitations and find them to be acceptable and require them to be incorporated in the license to manufacture. We consider this matter resolved.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (the Committee) completed a partial review of the application for a license to manufacture eight standardized floating nuclear plant units at the 188th meeting of December 4-6, 1975. A copy of the Committee's interim report dated December 10, 1975, is attached as Appendix B. Our response to these comments and recommendations are described in the following paragraphs.

(1) The Committee recommended that further consideration be given to methods for the assessment of probabilities for given accident events, such as those involving ships. The guidelines used by the staff in determining whether potential accidents in the vicinity of a site are to be considered as design basis events are discussed in Regulatory Guide 1.70.8 "Additional Information-Nearby Industrial, Transportation and Military Facilities". Design basis events external to a nuclear plant are defined as those accidents which have a probability of occurrence on the order of about 10^{-7} per year or greater and have consequences severe enough to affect the safety of the plant to the extent that 10 CFR Part 100 exposure guidelines could be exceeded.

For an applicant's consideration in determining design basis events, the staff has identified several accident categories with the categories being based upon the effect that a particular type of accident could have on a plant. The accident categories include explosions, flammable vapor clouds, toxic chemicals and fires. The probability of occurrence of each category from all potential hazard sources (transportation, industrial, military facilities) is considered in determining whether or not a particular category of accident need be considered a design basis event.

Usually in a site review there are several different kinds of hazardous material facilities or activities to be considered. The floating nuclear plant at an offshore site is unique in that the majority of the accident category hazards are related to shipping. However, for estuarine or riverine sites, industrial or military activities could well be the principal source of an accident category hazard. Thus the accident categories considered in determining whether or not an event will be considered a design basis event are the same as those considered for land based plants recognizing that for offshore sites shipping accidents will likely be the largest contributor for each of the accident categories.

The overall objective of the review in this area is to determine which accident effects, if any, should be included in the plant design. Determining that the probability of a type of accident, such as a ship accident, exceeds some guideline value does not in itself give the plant designer sufficient information. The designer also needs to know the potential effects on the plant produced by the accident. Thus, determining and specifying accidents in terms of accident categories which produce particular effects upon the plant has been the general approach followed in the review of the floating nuclear plant and is similar to the review for land based plants.

(2) The Committee indicated that they wished to be kept informed on the matters of containment shell buckling and the design basis tanker explosion. These matters are discussed in Section 3.8.1 and 2.10.2, respectively, of this Supplement.

(3) The Committee indicated that they wished to review the design and analysis of the emergency core cooling system and the upper head injection system. These systems are being evaluated by the staff in cooperation with the Committee on a generic basis with Westinghouse. The results of our evaluation will be implemented in our review of the floating nuclear plant design which incorporates systems of similar design. Our evaluation and conclusions will be included in a supplement to the Safety Evaluation Report.

(4) The Committee noted areas wherein it wishes to be kept informed. These areas included turbine-generator alignment, hull-coupled vibrations and stresses associated with platform towing operation. These matters are discussed in Sections 10.2 and 3.11 of this Supplement.

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(5) The Committee noted that it wished to be kept informed regarding the location and range of instruments for determining the nature and course of any accidents. This matter is discussed in Section 7.5 of this Supplement.

(6) The Committee indicated that it wished to be kept informed on the matter of verification of structural design of the floating nuclear plant. This matter is discussed in Section 3.11 of this Supplement.

(7) The Committee stated that consideration should be given in design to the possible provisions for redundant mooring systems. This matter is discussed in Section 2.6 of this Supplement.

(8) The Committee recommended further review of the design features that are intended to prevent the occurrence of fires and to minimize the consequences to safety-related equipment should a fire occur. This matter is discussed in Sections 8.4 and 9.5 of this Supplement.

(9) The Committee stated that it reserves judgment on the generic liquid pathway study that is currently being performed by the staff and applicant until it has had an opportunity to review and evaluate the relevant information. We will keep the Committee informed on this matter.

(10) The Committee suggested that analyses be made of any possible increases in the protection of public health and safety which may be obtained by an increase in containment design pressure. The staff is performing a study, as a part of its environmental review of the floating nuclear plant concept, to compare the environmental consequences of a large accident at a land-based reactor and a floating nuclear plant. The results of our study as appropriate, will be considered in our design requirements for the floating nuclear plant. We will keep the Committee informed on this matter.

(11) The Committee suggested that additional attention be given to means for protecting the critical wave and splash zone area of the platform where repair or renewal may not be practical under the anticipated operating conditions of the floating nuclear plant. This matter is discussed in Section 3.11.3 of this Supplement.

(12) The Committee stated that it believes that special consideration should be given to conformance with "as low as reasonably achievable" criteria. This matter is discussed in Section 12.0 of the Supplement.

(13) The Committee indicated that the review of the floating nuclear plant design for features that could reduce the possibility and consequences of sabotage should be continued. The staff considers the design conservatisms provided in the floating nuclear plant for protection against design basis accidents also reduce the chance that an act of sabotage could result in jeopardizing the health and safety of the public. However, the staff will continue to review the provisions for protection against sabotage in applications that utilize the floating nuclear plant design.

(14) The Committee recommended that further attention be given to the possibility of extended loss of offsite power due to natural events or other causes and the potential impact of this possibility on the requirements for emergency AC power. This matter is discussed in Section 8.2 of this Supplement.

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20.0 FINANCIAL QUALIFICATIONS

In the Safety Evaluation Report, we stated that we would report the results of our evaluation in a supplement to the Safety Evaluation Report. Our evaluation is presented below.

20.1 Introduction

The Nuclear Regulatory Commission's regulations relating to financial data and information required to establish financial qualifications for applicants for manufacturing licenses appear in Section 50.33(f) of 10 CFR Part 50 and Appendices C and M to 10 CFR Part 50.

The applicant, Offshore Power Systems, has applied for a license to manufacture eight floating nuclear power plants. The license is sought for a period of fourteen years beginning no earlier than January, 1977. No other Nuclear Regulatory Commission permits or licenses have been issued to or applied for by the applicant in connection with the manufacture of these plants. The purchasers of the plants are responsible for obtaining the necessary Nuclear Regulatory Commission construction permits and operating licenses. Assuming each purchaser obtains the necessary permits and licenses in a timely manner, plant commercial operation should follow completion of manufacture by no more than eighteen months.

Offshore Power Systems is an unincorporated joint venture of Westinghouse Electric Corporation and Westinghouse International Power Systems Company, Inc., Westinghouse International Power Systems Company, Inc., is a wholly-owned subsidiary of Westinghouse Electric Corporation. Westinghouse Electric Corporation owns 99 percent of Offshore Power Systems, and Westinghouse International Power Systems Company, Inc. owns the remaining 1 percent. An assessment of the financial qualifications of Offshore Power Systems to undertake the proposed manufacturing activity is essentially an assessment of the financial qualifications of Westinghouse Electric Corporation, since the one percent interest owned by Westinghouse International Power Systems Company, Inc. does not include an obligation to contribute capital to the venture.

Westinghouse Electric Corporation is a large, diversified enterprise and generally regarded as the second largest producer of electrical equipment in the world. Sales in 1974 amounted to $5,798.5 million, 35 percent of which was accounted for by the energy related product lines. Net income in 1974 was $28.1 million, down sharply from a high of $198.7 million in 1972. This significant reduction in net income was primarily the result of non-recurring losses experienced in the sale of its major appliances, mail order and record club businesses during 1974. In 1975, Westinghouse Electric Corporation had sales of $5,862.7 million and net income of $165.2 million, a substantial rebound from the abnormally low 1974 results.

20.2 Pricing Policy and Manufacturing Cost Estimates

20.2.1 Pricing Policy

The applicant has submitted a breakdown of the price of a floating nuclear power plant based on the May 1975 proposal to the Federal Energy Administration. The Federal Energy Administration proposal represents the most recent pricing policy for such a plant. The unit cost estimate has been itemized as follows:

	Unit Cost (dollars in millions)
Structures and improvements	$ 80.7
Reactor plant equipment	139.3
Turbine generator plant	129.2
Accessory electric equipment	39.0
Miscellaneous power plant equipment	15.4
Transmission facilities	10.5
Platform structures and specifically related systems	18.8
Testing (multi-systems)	2.1
Total Cost per Unit	$ 435.0

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The $435 million price per unit is a base price in January 1975 dollars, subject to escalation upward or downward as manufacturing costs fluctuate. The base price per unit will be escalated as follows:

Base Price	Escalation Index Employed	Escalation Base
55 percent labor	Average Hourly Earnings in Shipbuilding & Repairing Industry - Bureau of Labor Statistics	January, 1975
35 percent material	Steel Mill Product Index - Bureau of Labor Statistics	January, 1975
10 percent profit & overhead allowance	Gross National Product Implicit Price Deflator - U.S. Department of Commerce	First quarter, 1975

The escalation provision will enable Westinghouse Electric Corporation to maintain the financial integrity of the pricing policy in this venture. This is most important when one considers the potential impact future inflation could have on the manufacturing costs during this lengthy future period.

The manufacture of an individual floating nuclear power plant will not commence until an order has been ced for the plant. At present, the applicant has an order from the New Jersey Publ c rvice Electric and Gas Company f r four plants.

20.2.2 Manufacturing Cost Estimates d Sources of Funds

The estimated manufacturing costs (including the manufacturing facility) for the eight floating nuclear power plants is $3,287.5 million. The applicant submitted an itemization of the estimated manufacturing costs, including a detailed breakdown of the cost estimate of the manufacturing facility. This financial information was submitted with a request that it be accorded proprietary treatment. The staff reviewed the applicant's request pursuant to the provisions of 10 CFR 2.790. Based on this review, the staff concluded that the applicant's justification conformed to the criteria for proprietary treatment and, consequently, granted the request.

Through 1978, case requirements of $531.7 million will be provided from a continuation of internally generated funds ($443.8 million) and from funds provided by Westinghouse Electric Corporation ($87.9 million). After 1978, the cash requirements of $2,755.8 million will be provided by internally generated funds. The $87.9 million represents the maximum investment to be provided by Westinghouse Electric Corporation. The funds required to attain the maximum investment will be provided from the following sources:

Source of Funds	Amount in millions of dollars	Ratio
Sale of Interest Bearing Long-term Debt	$ 21.0	23.9 percent
Minority Intest	2.2	2.5
Preferred Stock	1.0	1.2
Common Stock	25.3	28.7
Internally Generated Funds	38.4	43.7
	$ 87.9	100.0 percent

The cash requirements generated by internally generated funds represent progress payments to be made by the purchasers. These progress payments will be in accordance with a payment schedule that is negotiated at the time of purchase.

Revenues from units sold are expected to cover the cost of manufacturing the units, amortization of the manufacturing facility, interest on money borrowed, and any other cost applicable to the project. While total cash requirements can be projected, any meaningful breakdown of the annual increments of such cash requirements must await firm information on the sale and delivery of the four floating nuclear power plants currently being marketed by the applicant.

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20.3 Conclusions

Based on the preceding analysis, which included the proprietary data, we have concluded that the applicant is financially qualified to manufacture the proposed eight floating nuclear power plants. Our conclusion is based on a determination that the applicant has reasonable assurance of obtaining the funds necessary to carry out the manufacturing activity for which the license is sought.

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21.0 CONCLUSIONS

In Section 21.0 of the Safety Evaluation Report, we stated that we would be able to make certain conclusions upon favorable resolution of the outstanding matters set forth in Section 1.10 of the Safety Evaluation Report. We have discussed these matters in this supplement and indicated a favorable resolution for each matter except for a single issue discussed on Section 1.10 of this supplement.

APPENDIX A

CONTINUATION OF THE CHRONOLOGY OF REGULATORY RADIOLOGICAL REVIEW OF FLOATING NUCLEAR PLANTS 1-8

September 18, 1975	Letter to Offshore Power Systems requesting additional information.
September 18, 1975	Letter from United States Coast Guard advising that Captain C. E. Mathieu has been transferred and that Commander John Deck III has taken his place.
September 25, 1975	Meeting with Offshore Power Systems to discuss generic liquid pathway study.
September 30, 1975	Letter to Offshore Power Systems summarizing results of July 15, 1975 meeting regarding environmental impact of postulated accidents associated with the NEPA.
October 3, 1975	Letter from Offshore Power Systems transmitting Final Report entitled "Evaluation of Hazards to a FNP from a Coastal Tanker Accident Near the Plant," dated September 30, 1975 with attached graph.
October 9, 1975	Meeting with Offshore Power Systems to discuss technical issues relative to the liquid pathway generic study.
October 9 and 10, 1975	Meeting with Offshore Power Systems to discuss implementation of generic liquid pathway study.
October 17, 1975	Letter from Offshore Power Systems transmitting Report No. SA-1000-14A96, "Results of Design Overspeed Turbine Missile Strike Probability Calculations on Vital Areas of the FNP Using the MIDAS Code."
October 20, 1975	Letter to Offshore Power Systems requesting additional financial information.
October 20, 1975	Letter from United States Coast Guard providing comments on United States Coast Guard Plan Lists.
October 20, 1975	Letter from United States Coast Guard providing comments on the Platform Hull Drydocking Equivalency document.
October 20, 1975	Amendment No. 19 provides additional information concerning safety related cooling water temp.
October 24, 1975	Meeting to discuss Offshore Power Systems proposal to increase site design envelope maximum basin water temperature from 85 to 95 degrees Fahrenheit (PDR Section 2.7.3).
October 29 and 30, 1975	ACRS Subcommittee Meeting.
November 4, 1975	Amendment No. 20 provides additional concerning shield building annulus.
November 7, 1975	ACRS full committee meeting.
November 10, 1975	Letter to Offshore Power Systems granting withholding of Control Rod Drive Mechanism analysis.

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November 10, 1975	Letter to Offshore Power Systems granting withholding of financial information.
November 13, 1975	Meeting with Offshore Power Systems to discuss generic liquid pathway study - discussion of liquid transport models.
December 1, 1975	Meeting with Offshore Power Systems to discuss design basis tanker explosion.
December 2, 1975	Meeting with Offshore Power Systems to discuss generic liquid pathway study.
December 3, 1975	Letter from Offshore Power Systems furnishing information concerning asymmetric loadings on reactor pressure vessel support.
December 9, 1975	Meeting with applicant to discuss generic liquid pathway study.
December 10, 1975	ACRS letter.
December 12, 1975	Letter to Offshore Power Systems transmitting ACRS letter.
December 12, 1975	Meeting with Offshore Power Systems to discuss basis tanker explosion.
December 17, 1975	Letter from Offshore Power Systems transmitting Offshore Power Systems Report No. RP-9991-16A50, "Operating Basis Wind for U. S. Atlantic and Gulf Coastal Locations."
December 18, 1975	Letter from Offshore Power Systems transmitting Offshore Power Systems report regarding hazards to a floating nuclear power plant from a coastal tanker accident near the plant.
December 18, 1975	Letter from Offshore Power Systems transmitting Offshore Power Systems report regarding containment shell buckling criteria.
December 18, 1975	Amendment No. 21 provides additional information concerning plant design report.
January 13, 1976	Letter from United States Coast Guard regarding fire tests of weirs for external fire protection.
January 16, 1976	Letter from Offshore Power Systems transmitting report regarding wind tunnel study of wind forces.
January 23, 1976	Letter from Offshore Power Systems transmitting report on design for air blast loading.
January 30, 1976	Letter from Offshore Power Systems transmitting report regarding hazards to a floating nuclear power plant from a coastal tanker accident near the plant.
February 5, 1976	Meeting with Offshore Power Systems to discuss technical issues relative to liquid transport modeling and scheduling problem.
February 17, 1976	Letter from United States Coast Guard regarding report on plant design.
February 23, 1976	Meeting with Offshore Power Systems to discuss containment shell buckling criteria and air blast loads resulting from the design basis tanker explosion.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 10, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON FLOATING NUCLEAR PLANT

Dear Mr. Anders:

During its 188th Meeting, December 4-6, 1975, the Advisory Committee on Reactor Safeguards completed a partial review of the application of Offshore Power Systems for a license to manufacture eight standardized Floating Nuclear Plant units in a shipyard-like facility located on Blount Island in Jacksonville, Florida. The Committee had previously reported to the Commission on its review of the concept of a Platform Mounted Nuclear Power Plant in its report of November 15, 1972. In addition, the Committee has had discussions of the Floating Nuclear Plant (FNP) concept in connection with the Atlantic Generating Station site review on which the Committee reported on October 18, 1973. The manufacturing facility site was visited on October 29, 1975 and the project was considered at a Subcommittee Meeting on October 29 and 30, 1975, in Jacksonville, Florida. The project was also considered during the 187th Meeting of the Committee in Washington, D. C., November 6-8, 1975. During its review, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff, the U. S. Coast Guard, and representatives and consultants of Offshore Power Systems. The Committee also had the benefit of the documents listed.

The FNP will make use of the Westinghouse RESAR-3 Consolidated Version four-loop pressurized water nuclear reactor having a core power output of 3411 MW(t). This reactor design is similar to that utilized at the Catawba Nuclear Station Units 1 and 2, reported on by the Committee in its report of November 13, 1973. The scope of the FNP design includes the Nuclear Steam Supply System (NSSS) and the Balance of Plant (BOP). The complete system, which is to be mounted on a large floating platform, represents a standard unit which is being designed for use at sites which fall within an envelope of parameters or specifications. The plant design includes specific requirements for major components, piping systems, and other information necessary to ensure that both the NSSS and BOP are designed to protect the system from site-related hazards. Application of the FNP concept will require an evaluation of each site to confirm its acceptability within the given envelope.

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With respect to the site envelope, the Committee recommends that further consideration be given to methods for the assessment of probabilities for given accident events, such as those involving ships. Rather than treat each potential accident situation as a separate class of event, it may be more appropriate in some cases to evaluate the significance of a given class of event on the basis of the total probability of all events within that class.

The NRC Staff has identified several issues which remain to be resolved. One pertains to the acceptability of criteria for containment shell buckling, including the behavior of the shell during construction. To be included in the assessment of this issue are the effects of deformation of the containment foundation. Another issue concerns the effects and consequences on the FNP of the explosion nearby of a petroleum tanker. These matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Evaluation of the Emergency Core Cooling System (ECCS) design in accordance with Appendix K of 10 CFR Part 50 is also an outstanding issue which has been identified by the NRC Staff. In this regard, the Committee has special interests relating to detailed assessments of the upper head injection system, the resolution of potential problems with the ice condenser pressure suppression system, and the margins available in the ECCS. The Committee wishes to review the design and analysis of both of these systems prior to the NRC issuance of a license to manufacture the FNP units.

In the course of its review, the Committee noted other areas wherein it wishes to be kept informed. These include any problems associated with turbine-generator alignment; hull-coupled vibrations (particularly as these relate to the potential of turbine failure and the generation of missiles); analysis of stresses on key components associated with platform towing operations; and the location and range of instruments for determining the nature and course of any accidents.

Since the FNP is a novel design requiring unusual structural reliability there is a need to develop plans for verification of structural design and to define the requirements for strain and deformation measurements, visual inspection during operational testing, and nondestructive inspection of critical FNP structures subsequent to operational loading. The Committee wishes to be kept informed.

Consideration should be given in design to the possible provisions for redundant mooring systems.

The Committee recommends that the NRC Staff and the Applicant review further the design features that are intended to prevent the occurrence of fires and to minimize the consequences to safety-related equipment should a fire occur. This evaluation should include a review of systems for detecting and protecting against fires, both within and outside the plant. This matter should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

Also to be evaluated are the consequences of, and any safeguards necessary to cope with, a major accident which could lead to the dispersal of a significant quantity of radioactive materials into the water surrounding the FNP. The Committee understands that this item is being evaluated by the NRC Staff and the Applicant. The Committee will reserve judgment on this item, which is both site and plant related, until it has had an opportunity to review and evaluate the relevant information.

The Committee suggests that analyses be made of any possible increases in the protection of public health and safety which may be obtained by an increase in containment design pressure.

The Applicant has suggested the use of a coating and a cathodic system to protect the platform against corrosion. The proposed cathodic system appears to be suitable for the underwater portion of the platform; however, additional attention should be given to means for protecting the critical wave and splash zone areas where repair or renewal may not be practical under the anticipated operating conditions of the FNP.

Because operating and maintenance personnel may be on board the floating platform for extended periods of time, and because shielding may be limited due to weight restrictions and limitations on available space, it is possible that doses and dose rates to personnel on the FNP may be greater than for land-based units. As a result, the Committee believes that special consideration should be given to conformance with the "as low as reasonably achievable" criterion.

The Committee believes that the Applicant and the NRC Staff should continue to review the FNP design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that further attention be given to the possibility of extended loss of off-site power due to natural events or other causes, and the potential impact of this possibility on the requirements for emergency AC power.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. The Committee believes that procedures should be developed to incorporate approved resolution of these items into the FNP.

The Advisory Committee on Reactor Safeguards believes, that subject to the foregoing and to other applicable matters discussed in its reports of November 15, 1972 and October 18, 1973, the Floating Nuclear Plant units can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public. The Committee will complete its review of this application when the necessary additional information has been developed.

Sincerely yours,

W Kerr

W. Kerr
Chairman

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References

1. Floating Nuclear Plant (FNP) Plant Design Report (PDR) Volumes 1-8

2. Amendments 1 through 17 to the PDR

3. Safety Evaluation Report by the Division of Reactor Licensing (DRL), dated September 30, 1975

4. Memorandum of Understanding Between the U. S. Coast Guard (USCG) and the U. S. Atomic Energy Commission for Regulation of Floating Nuclear Power Plants, dated January 4, 1974

5. Letter, dated September 5, 1975, Offshore Power Systems (OPS) to DRL, transmitting Westinghouse Report entitled "Valve Reliability, Turbine Inlet Valves," dated August 1975

6. Letter, dated September 2, 1975, USCG to DRL, providing comments on Offshore Power Systems Response to Staff Position Concerning Charpy V-Notch Testing of Weldments

7. Letter, dated August 25, 1975, OPS to DRL, providing additional turbine missile information

8. Letter, dated August 11, 1975, providing information on Emergency Core Cooling System Performance

9. Letter, dated August 8, 1975, OPS to DRL, transmitting Revision 1 to the Platform Hull Drydocking Equivalency document

10. Letter, dated June 3, 1975, OPS to DRL, providing information on turbine missile penetration of steel barriers

11. Letter, dated May 21, 1975, OPS to DRL, regarding external fire protection system

12. Letter, dated April 3, 1975, USCG to DRL, providing plan regarding details of exterior fire protection

13. Letter, dated March 10, 1975, USCG to DRL, providing comments on Platform Hull Corrosion Control Plan

14. Letter, dated January 30, 1975, OPS to DRL, regarding Damaged Platform Stability

References - Continued

15. Letter, dated January 23, 1975, OPS to DRL, transmitting report on Control Rod Drive Mechanism Analysis performed by Westinghouse
16. Letter, dated January 15, 1975, OPS to DRL, transmitting reports requested by USCG
17. Letter, dated January 3, 1975, USCG to DRL, providing comments on fracture toughness testing of hull steel for floating nuclear plants
18. Letter, dated December 17, 1974, OPS to DRL, transmitting errata sheet for report on external fire protection
19. Letter, dated December 5, 1974, OPS to DRL, regarding procedures for structural plans
20. Letter, dated November 22, 1974, OPS to DRL, transmitting revision to Equivalency Demonstration document
21. Letter, dated November 17, 1974, OPS to DRL, transmitting four reports referenced in the PDR
22. Letter, dated November 13, 1974, OPS to DRL, transmitting nonproprietary report on Emergency Trip Systems and Ultrasonic Inspection
23. Letter, dated November 12, 1974, OPS to DRL, transmitting document entitled "Floating Nuclear Plant Platform Hull Corrosion Control Plan
24. Letter, dated November 12, 1974, OPS to DRL, transmitting Westinghouse Electric Corporation reports
25. Letter, dated October 16, 1974, USCG to DRL, enclosing August 29, 1974 letter from OPS and USCG's October 11, 1974 letter to OPS
26. Letter, dated October 8, 1974, OPS to DRL, transmitting report entitled "Wind and Wave Persistence and Forecast Lead Times for Four Offshore Locations"

27. Letter, dated September 26, 1974, OPS to DRL, transmitting position on Anticipated Transients Without Scram

28. Letter, dated September 6, 1974, OPS to DRL, transmitting Westinghouse report entitled "A Comparison of Westinghouse Overspeed Protection to the Requirements of IEEE 279"

29. Letter, dated August 29, 1974, OPS to DRL, transmitting three Westinghouse reports

30. Letter, dated August 12, 1974, OPS to DRL, providing clarifying information regarding the calculation of transmission line reliability

31. Letter, dated August 8, 1974, OPS to DRL, transmitting MIDAS code report

32. Letter, dated August 7, 1974, OPS to DRL, transmitting revision to Emergency Power Equivalency document

33. Letter, dated August 2, 1974, OPS to DRL, transmitting Westinghouse Electric Corporation report on Analysis of the Probability of the Generation and Strike of Missiles From A Nuclear Turbine

34. Letter, dated July 2, 1974, OPS to DRL, transmitting document entitled "Platform Inclinations Due to Damage to Any One Side"

35. Letter, dated July 2, 1974, OPS to DRL, transmitting revision to Emergency Power Equivalency document

36. Letters, dated March 29 and February 25, 1974, USCG to DRL, regarding selection of hull material for Floating Nuclear Plants

37. Letter, dated January 7, 1974, OPS to DRL, transmitting information on Anticipated Transients Without Scram

38. Letter, dated October 26, 1973, OPS to DRL, providing interim information regarding Nuclear Plant Arrangement and Ice Condenser Design

APPENDIX C

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL DATA SERVICE
National Climatic Center
[redacted]
Asheville, N. C. 28801

Date : November 5, 1975

Reply to Attn. of: D5x1

To : Bob Kornasiewicz
Earl Markee

From : Harold L. Crutcher
Scientific Advisor

Harold L Crutcher

Subject: Fastest Mile 100-Year Return Estimate

Reference is made to our letter of March 10, 1975 signed by Mr. Bill Brower and to your visit here on November 4, 1975.

As indicated in our discussion, we see no need to revise our estimates of 160 and 360 mph for the extreme wind expected value and upper 0.975 probability confidence limit, respectively. These are for 100-year return values for anywhere along the coast from Corpus Christi, TX to Nantucket, MA and in the nearby oceanic areas any time.

It might be useful to stress the preliminary tables which we provided to you, which show that a 100-year return value has approximately a 1 in 3 chance of occurring in any 40-year period.

If in the course of future work, which hopefully would include more data, it becomes necessary to adjust the above estimates, you will be notified.

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