Safety Evaluation Report

NUREG-0054
(Suppl. No. 2 to NUREG-75/100)

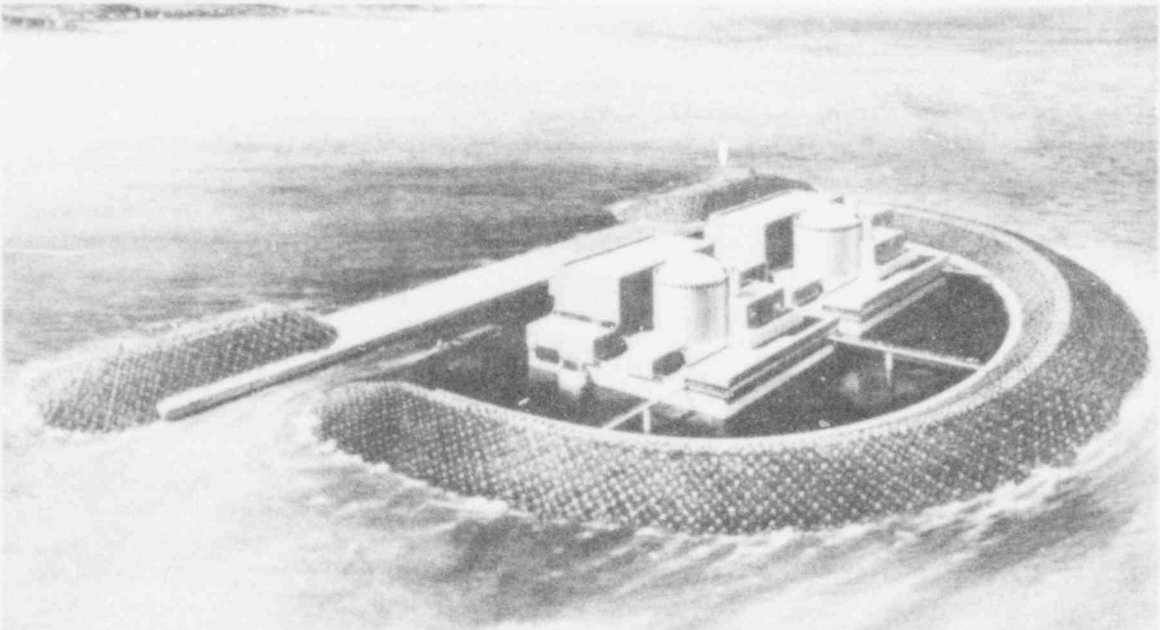
U. S. Nuclear Regulatory Commission

related to operation of
Offshore Power Systems Floating Nuclear Plants (1—8)

Office of Nuclear Reactor Regulation

Docket No. STN 50-437

October 8, 1976



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NUREG-0054
Supplement No. 2
to NUREG-75/100

SUPPLEMENT NO. 2
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

OCTOBER 8, 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, and Supplement No. 1 to the Safety Evaluation Report was issued on March 16, 1976.

The purpose of this supplement is to update the Safety Evaluation Report by providing (1) our evaluation of matters where our review of the information submitted by the applicant had not been completed when the Safety Evaluation Report and the first supplement were issued and (2) our responses to the comments made by the Advisory Committee on Reactor Safeguards in its report dated June 7, 1976.

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. Appendix C is a staff evaluation of potential accidents in the vicinity of a nuclear power plant. Appendix D is the Report of the Advisory Committee on Reactor Safeguards relating to the upper head injection system design. Appendix E is a staff letter to Westinghouse Electric Corporation discussing the ECCS-UHI evaluation model.

1.6 Site-Related Design Envelope

The site-related design envelope parameters as summarized in Table 1.2 (REVISED) in Supplement No. 1 to the Safety Evaluation Report is reproduced herein to reflect current design parameters utilized in the evaluation and typographical corrections. These changes are identified by a vertical margin bar.

1.10 Outstanding Issues

In Section 1.10 of Supplement No. 1 to the Safety Evaluation Report, we stated that all of the outstanding issues have been resolved with the single exception regarding our evaluation of emergency core cooling system design. This matter is discussed in Section 6.3 of this supplement.

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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 exccedance low spring tide plus drawdown produced by tsunami.	2.3

TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

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 spectra	(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 basis 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 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	3.7.1

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TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
	(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	95 degrees Fahrenheit	2.7.3
(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

TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(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) Precipitation 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 inch 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 inch 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 of the Plant Design Report	2.9
(13) Accident dose offsite must not exceed 10 CFR 100	Whole 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

TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

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) ≤ 2 foot from mean plane and ≤ 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 & 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 coincident 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

TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

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) Boat, 50,000 pounds, 29.3 feet per second (b) Wood plank, 4 inches x 12 inches x 12 feet, 420 feet per second (c) Steel pipe, 3 inch diameter, schedule 40, 10 feet long, 78 pounds, 210 feet per second (d) Steel rod, 1 inch diameter, 3 feet long, 8 pounds, 310 feet per second (e) Steel pipe, 6 inch diameter, 15 feet long, schedule 40, 285 pounds, 210 feet per second (f) Steel pipe, 12 inch diameter, 15 feet long, schedule 40, 743 pounds, 210 feet per second (g) Utility pole, 13.5 inch diameter, 35 feet long, 1,490 pounds, 210 feet per second (h) Automobile, 20 square feet frontal area, 4,000 pounds, 100 feet per second	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 (3150 long 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

TABLE 1.2 (REVISED) (Continued)

FLOATING NUCLEAR PLANT SITE DESIGN ENVELOPE

Requirement for Site Envelope Parameters	Envelope Parameter	Envelope Parameter Limit	Plant Design Report Section Reference
(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	
	(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.

2.0 PLANT-SITE INTERFACES

2.4 Wind and Wave Conditions

2.4.2 Wave Conditions

Wind and wave induced limiting motion design criteria are summarized in Table 2.1 of the Safety Evaluation Report. This table has been reproduced herein for clarification and to reflect typographical corrections which are identified by a vertical margin bar.

TABLE 2.1

WIND & WAVE INDUCED LIMITING MOTION DESIGN CRITERIA

SAFETY-RELATED STRUCTURES & EQUIPMENT

Design Motion**	Towing	Operating Basis	Design Basis
Equivalent Static Acceleration*** (in g's; gravitational acceleration, 32.2 feet per second per second)			
Longitudinal	0.34	0.05*	0.10
Transverse	0.34	0.05*	0.10
Vertical	0.42	0.05*	0.10
Plant Attitude			
Maximum Static Attitude	-	0.5 degrees	-
Maximum Attitude due to Static & Roll & Pitch	-	2 degrees	3 degrees

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1/ The significant wave height is defined as the wave height equivalent to the average of the upper one third of the waves in a typical wave train.

*operating basis 50 percent of design basis.

**at the four points shown on Fig. 3.7-4A of the Plant Design Report and based upon wave periods between 5 and 20 seconds.

***Equivalent static acceleration is defined on page 3.7-10(d) of the Plant Design Report.

2.10 Site Accidents

2.10.1 Site Envelope Criteria for External Accidents

2.10.1.1 Evaluation of Potential Accidents in the Vicinity of a Floating Nuclear Plant

We require that the floating nuclear plant be appropriately protected against events and conditions occurring external to the plant. The applicant has provided evaluations of potential hazards in the vicinity of a floating nuclear plant including consideration of shipping accidents. These evaluations were used to establish the site envelope parameters for external events which could cause unacceptable off-site radiological exposures.

In its review and evaluation, the staff has categorized or grouped potential accidents for purposes of determining if a general type of event is sufficiently likely that it must be considered in the design basis. Two basic categorization schemes may be employed. The first is an evaluation of the probability of accidents according to cause, such as railroad accidents or airplane crashes. The second is a grouping by potential effect on the plant such as explosive overpressures, concentrations of flammable or toxic chemicals, etc. There may exist a special situation where several events of a similar nature with respect to a potential plant impact from several causes may have a combined probability of occurrence in the range of less than one in a million per reactor per year, but individually none of which are so likely that they would clearly be included in the design basis. It is more likely that one or more type of events categorized by cause, such as a shipping accident, may be calculated to have a probability in the range of less than one in a million per reactor and therefore may need to be considered in the design. In practice, the staff reviews both situations.

Based on our evaluation (see Appendix C), we conclude that the staff's general procedure for judging whether external events need be included in the design basis is acceptable and consistent with our objective of assuring that the total probability of external events whose consequences exceed the dose guidelines of 10 CFR Part 100 is substantially less than one in a million per reactor per year.

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6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

We have evaluated the emergency core cooling system (ECCS) consisting of active and passive systems. The active components will consist of high head, medium and low pressure pumps that will be actuated by the safety injection signal. The passive systems will consist of an upper head injection (UHI) (single tank) and low pressure accumulator (four tanks) systems that are actuated when the reactor coolant pressure falls below preset values. The staff evaluation has included an extensive review of the Westinghouse emergency core cooling system upper head injection evaluation model which is the basis for the proposed floating nuclear plant design. The results of our generic evaluation are presented in the status report on the Westinghouse ECCS-UHI evaluation model dated August 13, 1976. This report identified several requirements for defining an acceptable evaluation model. Westinghouse is performing additional sensitivity studies to satisfy these requirements. A discussion of these sensitivity studies is presented in Appendix E.

The Advisory Committee on Reactor Safeguards (ACRS) in its independent assessment of this matter issued its report on the Westinghouse Electric Corporation's ECCS upper head injection evaluation model on September 14, 1976 (see Appendix D). The Committee stated that it believes that the Westinghouse Electric Corporation's ECCS UHI Evaluation Model, with the requirements set forth by the NRC staff, will conform to 10 CFR 50, Appendix K.

Offshore Power Systems will use the Westinghouse ECCS UHI Evaluation Model and has committed to include in the design of the ECCS any features or requirements set by the staff to conform with the requirements of 10 CFR 50.46. These features or requirements must be satisfactory to the staff prior to a decision on issuance of a license to manufacture.

The applicant submitted the results of a LOCA analysis for a double-ended cold leg guillotine break with a discharge coefficient of 0.6. The calculated peak clad temperature was determined to be below 2200°F (10 CFR 50.46 criterion); however, the evaluation model used in this analysis does not reflect all of the staff's requirements noted in the UHI/LOCA status report of August 13, 1976.

When an acceptable UHI/LOCA Evaluation Model is established, the applicant will submit appropriate additional LOCA analyses for his plant. This information must be reviewed prior to a decision on issuance of a license to manufacture. At that time, the staff would be able to make a final determination that the design was in conformance with 10 CFR 50.46.

Based on our evaluation of the information provided by the applicant, the generic evaluation of the Westinghouse ECCS-UHI evaluation model, the commitment by the applicant to include any features or requirements set by the staff and our requirement that this matter be satisfactorily completed prior to the issuance of a license to manufacture, we conclude that the design will be in conformance with 10 CFR 50, Appendix K requirements.

15.0 ACCIDENT ANALYSES

15.4 Radiological Consequences

15.4.3 Accident Releases to the Liquid Pathway

The Advisory Committee on Reactor Safeguards in its review indicated a concern relating to the consequences of an accident which could result in the release of radioactive materials into the liquid pathway. Responding to this concern, the staff and applicant initiated a study to determine whether the relative risks associated with accidental releases to the liquid pathway from water-based plants were different and significantly larger than those from land-based plants. The accidents included in the study ranged from minor operational incidents to very improbable core-melt events. The methodology used in the study is graphically shown in Figure 15.1. Consequences to man for all accidents were estimated in terms of radiation doses from drinking water ingestion, consumption of aquatic foods, and direct exposure from swimming and beach activities. Also considered in this evaluation was a very large accident (core-melt), one involving releases of substantial quantities of core inventory. For this event, the principal assessments were directed to radiation dose to man and fish, and long-term effects, such as genetic effects or complete species degradation.

The applicant provided estimates of the consequences of accidental releases from a floating nuclear plant for three typical ocean-based sites. The staff assessed consequences at four land-based plant cases and a floating nuclear plant estuarine site. The evaluation sites are shown on Figure 15.2. Details of the study are presented in the study report, NUREG-0140, "DRAFT LIQUID PATHWAY GENERIC STUDY," September 1976.

As part of the study a spectrum of accidents was considered. Six events were selected for detailed analysis. The estimated consequences of each event were calculated for each of the land-based and floating nuclear power plant sites. Table 15.1 summarizes the results of these analyses.

The estimated individual and population doses (assuming no steps are taken to mitigate the consequences) for the accident events A through E (see Table 15.1) are tablulated in Tables F-1 through F-25 of Appendix F of the study report. Evaluation of the data in Appendix F supports the following generalizations:

(1) For a given initiating event, the likelihood of a release to the liquid pathway is significantly less than a release to the gaseous pathway.

(2) The major exposure pathways are ingestion of drinking water and fish flesh for land-based plants and ingestion of fish flesh for ocean-based floating plants;

(3) The dose (total body and organs) to maximum individuals for these events range from less than 10^{-5} rem (about the dose received in one hour from the natural background radiation dose rate of 0.1 rem per year) to one rem, which is about twice the average annual radiation protection guideline value of 0.5 rem total body (or 1.5 rem thyroid) for people in unrestricted areas for normal operational releases (10 CFR Part 20);

(4) For the source terms given in Section 3.4 of the study report, radioiodine and radiocesium are the major contributors to the estimated exposures;

(5) Accident events A and B produce exposures comparable to those associated with normal operational effluent releases, i.e., doses to an individual of 10^{-3} rem (1 mrem) and population doses corresponding to about 1 man-rem; and

(6) The radioiodine releases as a result of accident events C and D produce the highest doses to an individual (thyroid) and population dose (thyroid man-rem).

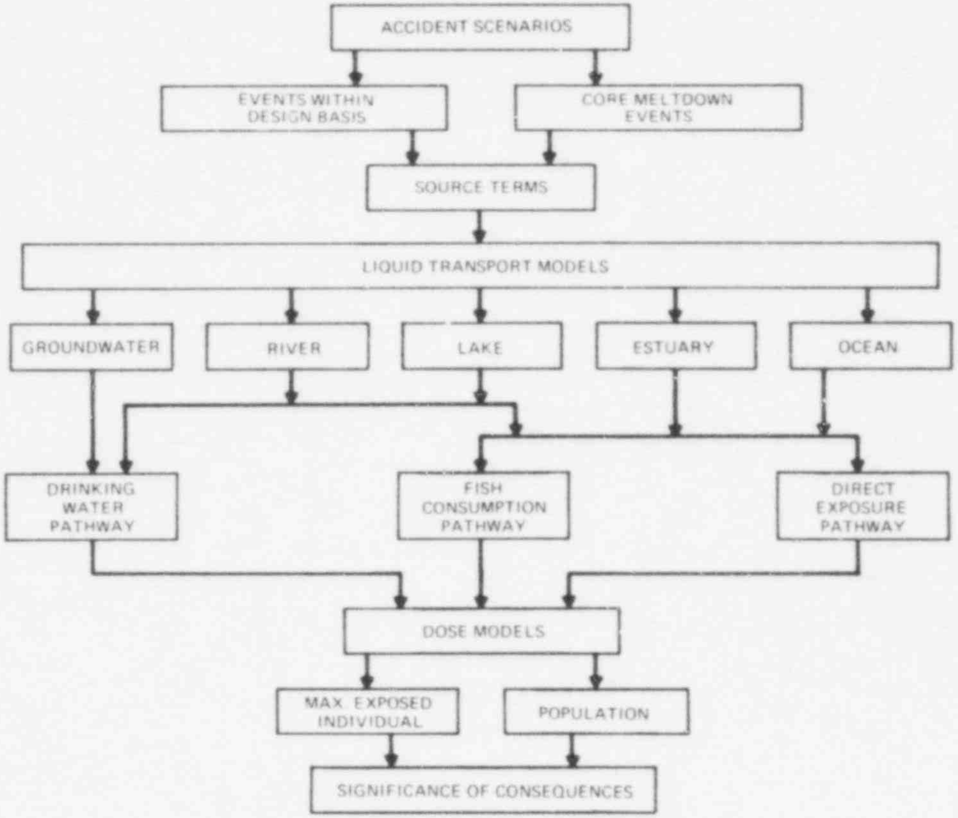

Figure 15.1 Liquid Pathway Generic Study.

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Figure 15.2 Classes of Sites Evaluated.

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Table 15.1 Summary of Liquid Pathway Results (Total Body Doses)

		DOSE TO MAXIMUM INDIVIDUAL (rem) / POPULATION DOSE (man-rem)					
	ACCIDENT EVENT* / SITE	A	B	C	D	E	CORE-MELT (man-rem)
LAND BASED PLANTS	RIVER	$<10^{-5}$ / <1	8.6×10^{-5} / 1.5	3.4×10^{-3} / 66	2.9×10^{-3} / 55	2.2×10^{-2} / 230	9.2×10^{4}
	LAKE	$<10^{-5}$ / <1	7.2×10^{-5} / <1	2.8×10^{-3} / 40	2.5×10^{-3} / 33	1.9×10^{-2} / 97	2.8×10^{4}
	DRY**	$<10^{-5}$ / <1	8.6×10^{-5} / 1.5	3.4×10^{-3} / 66	2.9×10^{-3} / 55	2.2×10^{-2} / 230	9.8×10^{5}
	ESTUARY	$<10^{-5}$ / <1	$<10^{-5}$ / <1	1.3×10^{-4} / 11	1.2×10^{-4} / 11	1.3×10^{-3} / 210	4.9×10^{3}
FLOATING NUCLEAR PLANTS	ESTUARY	$<10^{-5}$ / <1	$<10^{-5}$ / <1	1.3×10^{-4} / 11	1.2×10^{-4} / 11	1.3×10^{-3} / 210	$5\ \times10^{4}$
	OCEAN	$<10^{-5}$ / <1	6.9×10^{-5} / <1	2.6×10^{-3} / <1	1.9×10^{-3} / <1	5.5×10^{-3} / 120	$3\ \times10^{3}$

* Accident Event A – Radioactive Waste System Failures
Accident Event B – Releases as a Result of Steam Generator Leakage
Accident Event C – Loss-of-Coolant Accidents
Accident Event D – Rod Ejection Accident
Accident Event E – Accidents Involving Materials in Transit

** A dry site is characterized by the lack of nearby surface water. The radioactivity resulting from accident events A through [illegible] would be discharged to a liquid pathway in the same manner as routine discharges of liquid wastes. Therefore, the liquid pathway consequences for the dry site are determined by the character of the receiving water body. In the table for the dry site, the dose values for the river site are repeated, thus reflecting a maximum value of doses.

The floating nuclear plant sites evaluated were located in a typical estuary and at three typical ocean sites. As discussed in the study report, accidents at the various ocean sites are not anticipated to cause significantly different impacts. There is no significant difference in the predicated effects of credible liquid pathway accidents at floating or land-based plants in or adjacent to an estuary.

In making the comparison between the accident risks of a land-based pressurized water reactor and the floating nuclear plant, it became apparent that the significant difference between the designs (use of steel bulkheads instead of concrete for structures and the intimate contact of the plant with water) could affect the validity of prior judgments regarding the risks associated with very severe accidents, and specifically those accidents involving core meltdown and ultimate bulkhead meltthrough.

The prediction of the course of such accidents is a difficult and complex task because of the many and varied physical processes that could become involved. As with all complex accidents there is a spectrum of possible outcomes with results ranging from minor to severe.

Releases associated with accidents involving core melt may be represented by a long-term source due to leaching of radioactivity from core debris into water. Differences in the events leading to containment penetration and differences in the effect of debris contact with soil (or water) affect the estimated consequences as well as the effects of pathway differences.

For land-based facilities, the core-melt accident is postulated to result in releases which enter the groundwater system and are transported down-gradient to a surface water body enroute to water supply intakes or directly to private and public wells. The analyses were based on a one-year delay time before leaching could begin. The core debris at the floating nuclear plant was considered to enter the surface water body (ocean or estuary) directly - without delay - and to have much more surface water available for leaching than in a land-based plant. The results are also presented in Table 15.1. The analyses indicated that the consequences associated with the core-melt accident events at floating nuclear power reactors are generally comparable to the estimated consequences for the corresponding accidents at land-based plants.

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 its 194th meeting of June 3-5, 1976. A copy of the Committee's interim report dated June 7, 1976, is attached as Appendix B. Our response to the three remaining outstanding items is described in the following paragraphs:

(1) The Committee indicated that the adequacy of the emergency core cooling system should be reviewed by the NRC staff and the ACRS prior to issuance of a license to manufacture the FNP units. This matter is discussed in Section 6.3 of this supplement.

(2) The Committee indicated that a question still existed relating to the consequences of an accident which could result in the release of radioactive materials into the water. This matter is discussed in Section 15.4 of this supplement.

(3) The Committee recommended that further consideration be given to acceptable probabilities for each of several events, such as explosions in nearby ships, and that this consideration should be clarified by the staff. This matter is discussed in Section 2.10 of this supplement.

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 the remaining issues in this supplement and in Supplement No. 1 to the Safety Evaluation Report and have indicated a favorable resolution for each matter and therefore reaffirm our conclusion as stated in Section 21.0 of the Safety Evaluation Report.

APPENDIX A

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

February 11, 1976	Meeting with Offshore Power Systems to discuss programmatic issues relative to the Liquid Pathway Generic Study.
February 16, 1976	Letter from Offshore Power Systems regarding letter from United States Coast Guard, dated January 13, 1976, concerning planned fire tests of the falling water film system.
February 27, 1976	Letter from Offshore Power Systems providing information concerning steam line break inside containment.
February 27, 1976	Letter from Offshore Power Systems transmitting information relative to the plant design report.
March 2, 1976	Letter from Offshore Power Systems regarding submittal of information concerning the buckling criteria used in the design of the containment shell.
March 2, 1976	Letter from Offshore Power Systems transmitting Revision 2 to "Floating Nuclear Plant Platform Hull Drydocking Equivalency."
March 4, 1976	Letter from Offshore Power Systems providing information concerning 10 CFR Part 50, Appendix I.
March 16, 1976	Issuance of Supplement No. 1 to Safety Evaluation Report.
March 16, 1976	Letter from Offshore Power Systems transmitting proposed Manufacturing Conditions.
March 19, 1976	Meeting with Offshore Power Systems to discuss the generic liquid pathway study - radionuclide concentration and doses.
March 30, 1976	Submittal of Amendment No. 22 to Plant Design Report, consisting of information previously submitted, updated fluid system descriptions, revision to motion spectra for tornados and hurricanes, and other changes and errata.
March 31, 1976	Letter from Offshore Power Systems advising of tests to be conducted on the external fire protection system.
April 3, 1976	ACRS Subcommittee meeting with staff and Offshore Power Systems.
April 6, 1976	Meeting with Offshore Power Systems to discuss the results of Florida Power and Light Company's continuing study of a potential site for a floating nuclear plant at Cape Canaveral, Florida.

April 20, 1976	Meeting with Offshore Power Systems to discuss liquid pathway generic study - summary discussions.
May 4, 1976	Letter from Offshore Power Systems transmitting documents on the welding prequalification test program and associated test program.
May 10, 1976	Meeting with applicant to discuss liquid pathway generic study - interdiction techniques.
May 10, 1976	Letter to Offshore Power Systems advising of procedural changes relative to submittal of application and amendments.
May 24, 1976	Letter to Offshore Power Systems concerning generic review of Anticipated Transients Without Scram.
May 28, 1976	Letter from Offshore Power Systems transmitting minutes of meeting held May 25, 1976, concerning liquid pathway study.
June 7, 1976	ACRS interim report.
June 24, 1976	Letter from Offshore Power Systems transmitting "Consequences of Releases to Liquid Pathways," Topical Report No. TR 01A89.
July 15, 1976	Letter to Offshore Power Systems transmitting report by a consultant for the ACRS.
July 23, 1976	Letter from Offshore Power Systems providing response to request for additional information on liquid pathways.
July 23, 1976	Letter from Offshore Power Systems concerning United States Coast Guard comments on cathodic protection of Floating Nuclear Plants platform at the manufacturing facility.
July 23, 1976	Meeting with Westinghouse to discuss turbine generator design considerations for floating nuclear plants.
August 5, 1976	Letter to Offshore Power Systems transmitting report by a consultant for the ACRS.
August 6, 1976	Letter to Offshore Power Systems advising of changes to procedures for filing application amendments.
August 13, 1976	Letter from Westinghouse transmitting summary of Shippingport Rotor Bursting Incident and subsequent investigation.
August 20, 1976	Letter of Offshore Power Systems transmitting request for additional information concerning proposed Upper Head Injection System design.
September 7, 1976	Meeting with Offshore Power Systems to discuss Liquid Pathway Generic Study.
September 8, 1976	Letter from Offshore Power Systems providing information concerning the turbine generator.
September 9, 1976	Letter from Offshore Power Systems transmitting information regarding UHI system performance.

September 10, 1976	Submittal of Amendment No. 23 to Plant Design Report, consisting of information concerning ship collisions and other miscellaneous changes.
September 15, 1976	Letter to Offshore Power Systems transmitting request for additional information regarding Shippingport turbine failure.
September 23, 1976	Letter from Offshore Power Systems transmitting information regarding the Shippingport disc failure.
October 1, 1976	Letter from Offshore Power Systems transmitting information regarding ECCS.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 7, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON FLOATING NUCLEAR PLANT

Dear Mr. Rowden:

During its 194th Meeting, June 3-5, 1976, the Advisory Committee on Reactor Safeguards completed a partial review of the application of Offshore Power Systems (OPS) for a license to manufacture eight standardized Floating Nuclear Plant (FNP) units in a shipyard-like facility located on Blount Island in Jacksonville, Florida. This application was the subject of a Subcommittee meeting in Los Angeles, California, on April 3, 1976, as well as a number of earlier meetings with OPS (the Applicant) and with the Nuclear Regulatory Commission (NRC) Staff. The project was also considered during the 192nd and 193rd meetings of the Committee in Washington, D. C., April 8-10 and May 6-8, 1976, respectively. The Committee had most recently discussed this application in an interim report to the Commission on December 10, 1975. The Committee had earlier commented on the Platform Mounted Nuclear Power Plant in its report of November 15, 1972, and on the FNP concept in connection with the Atlantic Generating Station site in its report of October 18, 1973. During its review, the Committee had the benefit of discussions with the NRC Staff, the U. S. Coast Guard, and representatives and consultants of OPS. The Committee also had the benefit of the documents listed.

As noted in the Committee's Report of December 10, 1975, 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 MWt. This reactor design is similar to that utilized at the Catawba Nuclear Station Units 1 and 2, discussed 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 sytems, 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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Primary emphasis in this latest review of the FNP application was directed to an evaluation of progress being made on the resolution of issues raised by the Committee in its interim report of December 10, 1975. The review indicated that a number of the issues have been resolved. Those remaining are addressed below.

Evaluation of the adequacy of the emergency core cooling system (ECCS) design is an outstanding issue. This matter should be reviewed by the NRC Staff and the ACRS prior to issuance of a license to manufacture the FNP units.

A question which still exists relates to the consequences of an accident which could result in the release of radioactive materials into the water. The Committee wishes to withhold final judgment on the acceptability of the FNP application until the results on this question have been completed and have been evaluated.

In its most recent review, the Committee also gave further consideration to acceptable probabilities for each of several events, such as explosions in nearby ships, which could threaten the safety of the FNP. To assure that the sum of the probabilities of all such events will be acceptable, the Committee recommends that the specifications for this parameter within the proposed site envelope be suitably clarified by the NRC Staff.

The interim report issued by the Committee on December 10, 1975, listed a number of items on which it wished to be kept informed. The Committee recommends that the following items mentioned in that report be given additional attention. Resolution should be accomplished during the final design stages prior to completion of the construction of the first FNP unit. Issuance of a manufacturing license need not be contingent on the resolution of these items.

1. Independent analysis of containment shell buckling;
2. Turbine generator alignment and hull-coupled vibration;
3. Verification of structural behavior during towing operations;
4. Instruments to follow the course of an accident;
5. Fire protection design features;
6. Features to reduce the possibility and consequences of sabotage;
7. Possible increase in protection provided by an increase in containment design pressure;

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8. Possible plant modifications to protect against extended loss of offsite power.

Generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. The Committee repeats its earlier recommendation that procedures be developed to incorporate approved resolution of these items into the FNP units.

The Advisory Committee on Reactor Safeguards believes that, subject to the resolution of the outstanding issues and subject to the other matters discussed above, 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.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Additional Remarks by Messrs. M. Bender and S. H. Bush

The ACRS has always encouraged the examination of radionuclide dispersal into the environment for all types of accident circumstances, including a fully melted core that would penetrate containment. Such information is useful in understanding the ultimate seriousness of accidents and in determining the course of action that might be required should the totally unexpected ever occur. Nevertheless, a full-core melt that penetrates containment is not considered in the NRC's envelope of design-basis accidents. The frequency of occurrence of a core melt is expected to be well below that level at which substantial design changes are warranted. Additionally, we doubt that most design changes would ensure a substantive reduction in public health and safety risk attributable to such a nuclear accident.

It is our opinion that the FNP-ECCS, if properly engineered, will fully meet the requirements set forth in Appendix K of 10 CFR 50 and will adequately protect the plant against the possibility of a core melt. We do not believe, therefore, that the licensing of a Floating Nuclear Plant should hinge on the outcome of such studies.

We do believe the study of radionuclide pathways, resulting from a core melt, should be pursued and could properly include land-based as well as floating nuclear power stations. The results would be valuable in assessing the risk sensitivity of plant sites being considered for licensing and could be used as a site selection criterion when such marginal factors govern the benefit-cost basis for selecting siting alternatives.

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References:

1. Floating Nuclear Plant (FNP) Plant Design Report (PDR) Volumes 1-8
2. Amendments 1-21 to the PDR
3. Supplement No. 1 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation (NRR), dated March 16, 1976
4. Letter, dated December 3, 1975, Offshore Power Systems (OPS) to NRR, concerning asymmetric loadings on reactor pressure vessel supports
5. Letter, dated December 17, 1975, OPS to NRR, concerning operating basis wind for U. S. Atlantic and Gulf coastal locations
6. Letter, dated December 18, 1975, OPS to NRR, concerning hazards from a coastal tanker accident
7. Letter, dated December 18, 1975, OPS to NRR, concerning containment shell buckling criteria
8. Letter, dated January 13, 1976, United States Coast Guard to NRR, concerning fire tests for external fire protection
9. Letter, dated January 16, 1976, OPS to NRR, concerning wind tunnel study of wind forces
10. Letter, dated January 23, 1976, OPS to NRR, concerning design for air blast loading
11. Letter, dated January 30, 1976, OPS to NRR, concerning hazards from a coastal tanker accident
12. Letter, dated March 4, 1976, OPS to NRR, concerning conformance to 10 CFR 50 Appendix I
13. Letter, dated February 27, 1976, OPS to NRR, concerning steam line break
14. Letter, dated February 16, 1976, OPS to NRR, concerning testing of the falling water film system
15. Letter, dated March 2, 1976, OPS to NRR, concerning containment shell buckling criteria
16. Letter, dated March 2, 1976, OPS to NRR, concerning platform hull drydocking equivalency

APPENDIX C

EVALUATION OF POTENTIAL ACCIDENTS IN THE VICINITY OF A NUCLEAR POWER PLANT

1. Introduction

It has been a long-standing requirement that nuclear power plants be appropriately protected against events and conditions outside the nuclear power unit (see, for example, 10 CFR 50 Appendix A Criterion 4). Consequently, it is required that evaluations be performed of potential hazards in the vicinity of a nuclear power plant. Typically this includes consideration of traffic accidents, such as barge groundings and airplane crashes, and nearby industrial activities such as munitions or chemical factories.

The purpose of such evaluations is to determine whether the probability of external events, which subsequently cause unacceptable offsite exposures, are excessive, and if they are, to determine what provisions are needed in the power plant to reduce risks to an acceptable level. Logical questions are what constitutes an acceptable level of risk and how does the staff's review provide reasonable assurance that this risk level will not be significantly exceeded.

These questions are discussed below. A rationale for the general risk criteria is developed. Specific issues such as that raised by the ACRS in the reports on floating nuclear plants are discussed (the ACRS issue dealt with acceptable probabilities for each of several events, such as explosions in nearby ships, which could threaten the safety of the FNP).[1]

2. General Safety Objective

Criteria for considering acceptable risks from external hazards were developed so as to be compatible with the general objectives of the Commission rules and practices to assure that no undue risks to the health and safety of the public result from plant operation. In substance, the staff's reviews of external events and potential plant accidents aim first at determining whether a given event should be considered in the design bases and second, what measures are appropriate to protect against those events which are to be accommodated.

In establishing the boundary between accident sequences that are to be within the design basis envelope (for which engineered safety features are provided), and those for which no further protective features are considered necessary, the NRC staff used the safety objective that the risk to the public from all reactor accidents should be very small compared to other risks of life. That is, the incremental burden to society from this mode of power generation should be small.

One basis for the staff's objective stems from the philosophy that was expressed in submitting the draft version of 10 CFR 100 for public comment:

> The objective of these guides and of all Commission activities involving reactor licensing and operation is to keep the exposure of individuals to radiation at a minimum in the event, however unlikely, that an accident should occur with a reactor.[2]

Perhaps a more complete understanding of the basis for the staff's general objective can be found in an early ACRS review of siting criteria, where it was stated:

> Incidentally, we reject, as premature, the concept that damage to people from reactor accidents be no greater than that accepted in other industries, although in the future this might become a guiding principle The reasons for this rejection are two fold. We do not have sufficient information on the probability of accidents to make use of this concept in site evaluations. We do use, of course, the fact that the probability of a serious accident is very low. Secondly, we recognize that the atomic power business has not yet reached the status of supplying an economic need in a manner similar to that of more mature industries; and, therefore, arguments of taking customary risks for the greater good of the public are somewhat weak. At the same time, we do not want to imply that the restrictions placed on site locations during the development life of atomic power will necessarily be carried over to the period of maturity of the atomic power industry.[3]

As of August 19, 1974,[4] the Commission's policy has been to assure that accident risks are low relative to other risks of life - that something less than "customary risks" should be obtained.

3. Quantitative Risk Criteria

In attempting to deal with events that have a very low likelihood of occurrence, there is difficulty in establishing a value for the likelihood of occurrence that can be "proved." We have, in the United States, statistics that give reliable numbers for the likelihood of an individual being involved in an automobile accident in any given year. This is because there are, unfortunately, a very large number of such accidents each year and the statistical base for the probability determination is more than sufficient for the purpose. However, for postulated events of very low likelihood, that might occur only once in a thousand or more years, and have not occurred thus far, there is simply no base of experience from which to obtain probabilities in a direct way. This is the situation with regard to accidents that fall outside the design basis envelope for a nuclear plant. Such accidents may involve sequences of failures, each one of which is, in itself, relatively unlikely. The determination of the probabilities of accidents more severe than the spectrum of design basis accidents, then, is necessarily a matter of judgment. Applicable experience with various components and systems similar to those in nuclear plants provides some guidance in making these judgments, where such components and systems exist in other areas of industrial technology. The NRC staff uses such experience, together with its best judgment based on all relevant experience, in nuclear plants and elsewhere, in estimating the likelihood of significant accidents and in determining whether they should be within the design basis envelope or, alternatively, are sufficiently remote in likelihood so as to be an acceptable risk.

One general objective used to guide decisions in this area is that there be no greater than one chance in one million per year for potential consequences greater than the 10 CFR 100 dose guidelines for an individual plant. Some implementation actions relating to this objective are discussed below.

While this is a quantitative risk criterion only in a very limited sense, it does provide a useful benchmark in considering whether a given event or type of event of potentially serious consequences is sufficiently likely that accommodation is required.

A risk acceptance curve, along the lines of the probability-consequence estimates in WASH-1400[5] would provide a more rigorous criterion, and the staff is exploring possibilities of this type. However, because significant elements of the staff's review are deterministic rather than probabilistic,[6] the aforementioned general objective is likely to be retained in the immediate future.

4. General Safety Objectives for External Hazards

Working towards an objective of one chance in a million per reactor per year for potential consequences greater than Part 100 guidelines requires that any single event of this type have a probability objective of less than one in one million such that the sum of probabilities for all types of events is about 1 in 10^{-6}. A simple reading of Regulatory Guide 1.70 (the information required to present and assess in safety analysis reports) and NUREG-75/087 (the Standard Review Plan) discloses a requirement to consider the following types of events:

Internal Events

Loss-of-coolant accidents
Reactivity accidents (such as rod ejection)
Steam generator tube ruptures
Steam line breaks
Fuel handling accidents
Anticipated transients without scram
Turbine missiles

External Events

Earthquakes
Floods
Tornadoes
Airplane crashes
Explosions
Flammable vapor clouds

Toxic chemical release
Fires
Collisions with intake structures
Liquid spills (onto water) drawn into intake structures

Some of these events (or results of events) are required to be accommodated in design. For such events there is likely to be some residuum of risk that will be more severe than postulated for purposes of design (i.e., due to the occurrence of worse than design parameters or because of degraded performance of equipment assumed to operate properly in the design analysis).

Other events required to be considered may be dismissed because they are so unlikely as to warrant no special provisions in design.

There are 16 events listed above. Assuming no other contributors to this list, each event would have to be controlled to or allocated about 6×10^{-8} per year ($10^{-6}/16$), if risk were to be uniformly distributed amongst them.

As a practical matter, all events with the potential for having consequences in excess of 10 CFR 100 guidelines will not be of equal probability. For reasons noted in 1 above, in some cases it is not possible to obtain good estimates of event probability; mathematically equal allocation would not easily treat this situation.

Similarly, all events do not have the same potential consequences. For example, one might postulate an event involving a tube rupture and excessive iodine spiking. While such an event could exceed 10 CFR 100 it would simply not be the same degree of problem as an event involving core melting. While this example may be extreme it does indicate the need to consider the likely continuum of potential consequences above the 10 CFR 100 threshold. WASH-1400 presents a more complete survey of the range of consequences that one might associate wtih events in the 10^{-6} per year and less category, although it too is an ensemble of point analyses (with 9 PWR consequence categories).

WASH-1400 also presents the results of the first detailed attempt at developing an ensemble of point analyses both in terms of probability and consequences. Many reviewers correctly note that alternate ensembles of point analyses may be developed (with some groups suggesting that "truth" lies in a general bias towards greater consequence or probability than set forth in WASH-1400 and others noting the reverse).

An alternate view may equally be held, namely that all views have merit and value if viewed with an aim towards estimating the range of uncertainty (variability) one might associate with various sets of probability and consequence analyses.

The point of the foregoing is to indicate the difficult and perhaps futile nature of attempts to rigorously estimate risk (in terms of its components or probability and consequences) except by comparison with other perceived risks. The general safety objective set forth in 3 above is precisely of this form.

The staff, as noted in Regulatory Guide 1.70 Section 2.2.3 and Standard Review Plan 2.2.3 approached certain risks stemming from external human activities in a similar and derivative fashion.

Recognizing that there may be a number of events whose potential consequences may exceed 10 CFR 100 guidelines, a general objective has been to consider in the design, external hazards whose individual probability is of the order of 10^{-7} or more. This objective is seen as compatible with the general safety objective if the probability of all such events taken together is substantially less than 10^{-6} per reactor per year (so that the combination of the "16" plant accidents and external hazards will not exceed 10^{-6} per year).

5. Implementation of the Safety Objective

In the ACRS deliberations on Offshore Power Systems application floating nuclear power plants, concern was expressed regarding the means by which the staff's review was sufficient to assure that the sum of the probabilities of events such as explosions from nearby ships will be acceptable.[1] Discussed above is the basis for the staff's general objective that the probability of all external hazards be substantially less than 10^{-6} per reactor year. The means by which the staff implements this general objective are outlined in two documents, as follows:

From R.G. 1.70:

> 2.2.3.1 Determination of Design Basis Events - Design basis events external to the nuclear plant are defined as those accidents that have a probability of occurrence on the order of about 10^{-7} per year or greater and have potential consequences serious enough to affect the safety of the plant to the extent that Part 100 guidelines could be exceeded. The determination of the probability of occurrence of potential accidents should be based on an analysis of the available statistical data on the frequency of occurrence for the type of accident under consideration and on the transportation accident rates for the mode of transportation used to carry the hazardous material. If the probability of such an accident is on the order of 10^{-7} per year or greater, the accident should be considered a design basis event, and a detailed analysis of the effects of the accident on the plant's safety-related structures and components should be provided.

and from SRP 2.2.3:

> II. Acceptance Criteria - The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines exceeds the NRC Staff objective of approximately 10^{-7} per year. The methods of calculating the radiological exposures resulting from these events are acceptable if they are consistent with methods used for calculation of other accident radiological exposures (e.g., Standard Review Plan 15.6.5). Because of the difficulty of assigning precise numerical values to the probability of occurrence of the types of potential hazards generallly considered in this review plan, judgment must be used as to the acceptability of the overall risk presented by an event.
>
> In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be estimated using assumptions that are as realistic as is practicable. In addition, because of the low probability events under consideration, valid statistical data are often not available to permit accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.
>
> The effects of design basis have been appropriately considered if analyses of the effects of those accidents on the safety-related features of the plant have been performed and appropriate measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.

The analyses performed by the staff and applicants are conservative, even when attempts are made to develop "realistic estimates" in conformance with the above criteria. For example it is common to equate impact with serious (greater than 10 CFR 100) consequences. It is certain that not all external events, even if realized, would cause serious consequences. A crash of an airplane large enough to potentially cause failure of vital structures is assumed, for purposes of such analyses, to cause serious consequences. In fact, an airplane may well strike a glancing blow and not cause other than economic loss. This factor is not credited.

Therefore, should an estimate of the probability of a given accident fall in the range of 10^{-6} to 10^{-7} there is some assurance that the real probability is a significantly smaller value. Similarly, when a criterion of 10^{-7} is used for each of several events, in concert with usual staff analysis practices, this constitutes in effect a real probability criterion considerably more restrictive than that number of events times 10^{-7}. For example, in the case of the FNP, four event categories were selected with a cumulative probability criterion of 4×10^{-7}. For example, in the case of the FNP, four event categories were selected with a cumulative probability criterion of 4×10^{-7}. In effect, it can be argued that the criterion as implemented is substantially less.

A further case in point can be found in recent evaluation of airplane crash risks in the Hartsville review and barge traffic in the Beaver Valley review. In both cases, careful reviews were made and it was found that the estimates based on usual staff assumptions were conservative by about an order of magnitude.

For an applicant's considerations 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.

In reviewing a specific location for a Floating Nuclear Plant the staff has to consider, for example, such items as impact with a munitions ship which could theoretically cause serious structural damage to the safety systems of a nuclear power reactor. Since, considering costs, it is not practical to build a nuclear plant to withstand a nearby impact of a munitions ship, the staff has to ascertain, based on a specific location the barge or ship traffic that travels near the site, and determine statistically that the likelihood of such an incident on the seawall is extremely unlikely and on the order of 10^{-7} per year. For this reason, it is unlikely that a Floating Nuclear Power Plant can be located in the near vicinity of a shipping lane used to transport munitions. In evaluating a specific site for a Floating Nuclear Power Plant, it should be noted that we are not faced with a situation where all accident impacts exist at a single site. During the past several years, for example, the staff has determined a number of man-made hazards to which several specific plants had to be designed to withstand. In Beaver Valley, for example, the applicant was required to move a natural gas line from a distance of closest approach of 500 feet to 1000 feet to prevent the reactor containment building from being engulfed in a flammable gas mixture. In addition, at this site the intake structure had to be designed to withstand the impact and explosion of a gasoline barge. In this latter instance, it was not feasible to modify the existing structure and therefore the applicant chose to provide an alternate emergency intake to provide cooling water to safely shut down the plant following a loss of the original intake structure. At Three Mile Island near Harrisburg, Pennsylvania the applicant was required to design the contaiment structure and all engineered safety systems required for safe shutdown to withstand the impact of a large commercial aircraft. Other facilities such as Zion and Montague were designed to take the impact of light aircraft, since statistics indicated that large commercial aircraft would not be utilizing the airspace at a low altitude over these facilities. Another example is the Brunswick plant located on the Cape Fear River where munitions could be shipped to within two miles of the site. The staff concluded that there was not likely to be any adverse effect on the plant as a result of the operations at the munitions terminal. Experience to date has therefore shown that one or possibly two man-made hazards may exist at a specific site but there have been no sites licensed to date which are designed for a series of manmade hazards. This is not a mere coincidence. The operationally convenient tools of the Standard Review Plan acceptance criteria permit a fairly rapid and orderly screening of events which are sufficiently likely that they need to be considered very carefully in the staff review and, if appropriate, accommodated in design. As discussed above, the staff's objectives are to assure that the overall risks associated with external events are acceptably low. If the staff's review indicates that a number of potential hazards had significant or marginallly acceptable risks then a separate finding would be required that the sum of these risks was acceptable.

In the case of the FNP the set of risks unique to the selected site are not yet characterized (except through the separate applications). Consequently, to indicate the general controls on external risks, a matrix table was devised (Figure 1). If each were precisely probable enough (to 9.99×10^{-8}) then, in theory the sum would be 4×10^{-7}. This is not an unacceptable value given that the other elements of risk (the initiating events listed in 4 above) will certainly be below the 10^{-7} value. For reasons such as this, such a situation as shown in the matrix table is obtainable only in theory, not in practice. Consequently, it is not inconsistent with achieving the general safety objective of one in one million per plant per year.

6. Summary and Conclusions

In summary, each potential external accident may be treated as a separate event for purposes of determining which must be considered in the design basis. There may exist a peculiar situation where several events of a similar nature with respect to potential plant impacts may have a probability of occurrence in the range of 10^{-7} or less, but none of which are so likely they would clearly be included in the design basis. A more likely situation is that one or more events may be calculated to have a probability in the range of 10^{-7} and therefore may be considered marginal.

Based on the staff's experience, it is believed that no situation will occur wherein more than four such events may co-exist at a single site. Further, the analyses conducted by the staff, even when done "realistically" contain known elements of conservatism. Considering this in the light of the staff's objective of assuring that the total probability of external events whose consequences exceed the dose guidelines of 10 CFR 100 is substantially less than one in a million per reactor per year, the matrix format of FNP (Figure 1) is judged acceptable.

7. References

1. Letter, Dade W. Moeller, Chairman, ACRS to Marcus A. Rowden, Chairman, NRC, "Interim Report on Floating Nuclear Plant," June 7, 1976.

2. AEC Announcement, "AEC Issues Reactor Site Criteria Guides for Public Comment," February 10, 1961.

3. Letter, ACRS to John A. McCone, "Reactor Site Criteria," August 28, 1960.

4. Atomic Energy Commission, "Protection Against Accidents in Nuclear Power Reactors, Interim General Statement of Policy," effective August 21, 1974.

5. "Reactor Safety Study, An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, October 1975 (Figures 6-1 through 6-3).

6. "Current Plans of the Regulatory Staff for the Use of Probabilistic Assessment," paper by D. G. Eisenhut and R. C. DeYoung presented at American Nuclear Society Winter Meeting, November 17, 1976.

8.0 Some Observations on Related Matters

8.1 Risks to Individuals

The general safety objective and its derivations relative to external hazards are stated in terms of risks per reactor. An individual close to a power plant site boundary would have greater relative risks than an individual located at some more removed distance. For all "credible" accidents (those required to be within the design basis), this factor is controlled only by limiting the magnitude of the consequences to any individual. This, of course, is through dose limits for 2 hour exposures at the site boundary and through provisions for protective measures. No explicit requirement exists to develop emergency plans that explicitly provided preferential action for individuals close to the plant, although such a policy is implicitly affected.

8.2 Risk to the General Population

The notion that risks to the population at large should be controlled has been considered since the formative stage of 10 CFR 100. The use of a low population zone and population center distance were developed as the control index. The use of specific man-rem limit has been considered for over a decade.

The WASH-1400 results confirm that accident risks are roughly proportioned to population density. The staff has a population density screening system; although no firm policy has been expressed, it is generally acknowledged that sites "worse" than the envelope of site population densities at Newbold Island, etc., would not be acceptable.

An acceptable man-rem dose for events within the design basis could be back-calculated, based on currently acceptable practice. Acceptable man-rem doses have not been developed for events beyond the design basis. As in other severe situations, emphasis is placed on the capability to effect protective measures, such as evaluation, so as to avoid impacts of a disastrous nature.

8.3 Risks at a Site

As noted in 4.1, risk objectives are generally stated in terms of per reactor year. Consequently the cumulative risks to a local population is that associated with all the reactors at a site. A site with a quad unit could be permitted, theoretically, to expose the surrounding community to four times the risks as those from a single unit generating station. Two obvious rationales for such a practice can be agreed: (1) the risk is still small relative to other risks of life and therefore, the fundamental objective is met and (2) at such low individual risk levels, general societal impact is the pertinent criterion in which case the cumulative risk from all operating reactors to the population as a whole is the relevant consideration.

8.4 The Relevance of a Low Risk Criterion to External Events

As noted in the preceding discussions, the staff has implicitly allocated a fraction of the overall risks associated with power plant accidents to external causes. It should be kept in mind that the risks to the public may be governed more by the external event itself than by the impact of the event on a power plant which in turn has an impact on the public. "Common" disasters such as a major earthquake or a major flood (as discussed in WASH-1400) are in this category.

It has also been concluded by some, as in the Reactor Safety Study, that "external events are not expected to have a major impact on the risks associated with reactors" (p. 172, Main Report). If this conclusion is correct, the cost-effectiveness of provisions to protect against marginal external risks (in the range of 10^{-7}) is questionable.

On the other hand, it may be equally argued that the conclusion is valid only because of measures already taken in the design to reduce other accident risks (such as concrete containment and separation of redundant safety related equipment). A substantial reduction in protection from one class of external events may not, of itself, be of any significance to overall risks associated with reactors but may in turn cause the risks of other events to be of relatively greater significance. To illustrate, contemporary structural designs are controlled in strength in consideration of tornado missiles. Should more detailed reviews of missile risks result in a substantial downward shift in the spectrum of missile velocities to be considered, structural thickness could be reduced.

Airplane crash probabilities are generally of the order of 10^{-5} per year. With current structural thickness, the likelihood of a <u>damaging</u> impact is generally very small (less than 10^{-7} per year). If the structural requirements were substantially relaxed, the probability of a damaging impact may begin to approach that of the overall crash probability and hence become a significant contributor to risk.

The point of the above is merely to illustrate that individual risks may be considered separately, but any action should be made with a view of the impact of such an action on indirectly related risks.

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LIMIT FOR EVENTS NOT IN DESIGN BASIS (PROBABILITY PER YEAR)

ACCIDENT EFFECT ACCIDENT SOURCE	EXPLOSIVE OVERPRESSURE	FLAMMABLE VAPOR CLOUD	TOXIC CHEMICAL CLOUD	MISSLE IMPACT ON PLANT STRUCTURE	MAXIMUM PROBABILITY FROM ACCIDENT SOURCE TYPE
SHIPPING	10^{-7}	10^{-7}	10^{-7}	-	3×10^{-7}
TRUCKS	10^{-7}	10^{-7}	10^{-7}	10^{-7}	4×10^{-7}
RAIL	10^{-7}	10^{-7}	10^{-7}	-	3×10^{-7}
AIRCRAFT	-	-	-	10^{-7}	10^{-7}
FIXED STORAGE FACILITY	10^{-7}	10^{-7}	10^{-7}	-	3×10^{-7}
ALLOWABLE PROBABILITY FOR AN ACCIDENT EFFECT	10^{-7}	10^{-7}	10^{-7}	10^{-7}	

TOTAL PROBABILITY ALLOWANCE FOR SITE RELATED ACCIDENTS (4 TYPES) 4×10^{-7}

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 14, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON WESTINGHOUSE ELECTRIC CORPORATION'S ECCS UPPER HEAD INJECTION EVALUATION MODEL

Dear Mr. Rowden:

At its 197th meeting, September 9-11, 1976, the Advisory Committee on Reactor Safeguards completed a review of the Westinghouse Electric Corporation's ECCS Upper Head Injection (UHI) Evaluation Model. Six Subcommittee meetings have been held with representatives of the Westinghouse Electric Corporation and the Nuclear Regulatory Commission, the first being held at Monroeville, Pennsylvania, on June 25, 1975, and the remaining meetings at Washington, DC, on October 4 and December 20, 1975, and March 16, June 15, and September 2, 1976. The Committee also had the benefit of the documents listed below.

The NRC Staff has taken into account the special features of UHI and the supporting research and development in order to formulate evaluation model requirements which suitably conform to 10 CFR 50, Appendix K. Further, sensitivity studies are being performed by Westinghouse to provide assurance that sufficiently conservative bounds for the evaluations will be included.

The ACRS believes that the Westinghouse Electric Corporation's ECCS UHI Evaluation Model, with the requirements set by the NRC Staff, will conform to 10 CFR 50, Appendix K. The approved evaluation model, with appropriate use of plant-specific parameters on a case-by-case basis, will aid in the licensing reviews.

The ACRS encourages the NRC Staff to continue its deliberations in seeking accelerated and coordinated programs for establishing meaningful experimental facilities and independent analytical tools for studying the performance of UHI-ECC systems.

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As noted in its November 20, 1974 Report on Evaluation Models for Commission Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, the ACRS "remains mindful that the Evaluation Models, in themselves are not the desired end products, but that effective, reliable emergency core cooling systems are the objective." Continuing efforts are needed for implementing the safety research programs to provide bases for confirming design margins and for improving ECCS reliability and capability.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES:

1. "Status Report to the Advisory Committee on Reactor Safeguards in the Matter of Westinghouse Electric Corporation ECCS-Upper Head Injection Evaluation Model Conformance to 10 CFR 50, Appendix K," Proprietary (August 1976) U.S. Nuclear Regulatory Commission - Office of Nuclear Reactor Regulation.
2. WCAP-8400 - "ECCS Heat Transfer Experiences with Upper Head Injection," Proprietary, Volume I (October 1974), Volume II (January 1975), Volume III (August 1976), Westinghouse Electric Corporation.
3. WCAP-8479, Revision 1 - "Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection," Proprietary (August 1976) Westinghouse Electric Corporation.
4. WCAP-8582 - "Blowdown Experiments with Upper Head Injection in G-2, 17X17 Rod Array," Proprietary, Volume I (January 1976), Volume II (August 1976), Westinghouse Electric Corporation.
5. WCAP-8793 - "G-2, 17X17 Refill Heat Transfer Tests and Analysis," Proprietary (August 1976) Westinghouse Electric Corporation.

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APPENDIX E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 4 1976

Mr. Clement Eicheldinger, Manager
Safety and Licensing
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Eicheldinger:

On August 13, 1976 we published a status report on the Westinghouse ECCS UHI evaluation model. A number of unresolved issues were identified in that status report. At the ACRS ECCS subcommittee meeting on September 2, four of these issues were characterized as not having well-defined plans for resolution. These were:

1.) Drift Flux Model in the core

2.) Core Flow Behavior

3.) Unquench during counter current flow

4.) Upper head temperature.

A number of other issues were identified as having solution procedures well-defined. These other issues are not expected to present any significant difficulty but must be considered and resolved before our final model approval.

In order to provide a basis for resolution of the issues described above a plan was developed. This letter is to document our understanding of that agreement. An enclosure to this letter describes the details of how each issue should be addressed to effect a resolution. Sections 1.0, 2.0, and 3.0 of the attachment address the four major items. Sections 4.0 through 7.0 discuss additional items needed for selection of a model. Section 8.0 identifies items which must be addressed before approval of a final model. Westinghouse should provide a schedule for completion of these items.

The primary emphasis is on sensitivity studies needed to address the first 3 open issues. The sensitivity studies outlined are somewhat involved. Therefore, some guidelines are necessary to assure an orderly and meaningful resolution. Each study should proceed carefully. Time to analyze the results at appropriate points along the way should be provided. This should occur at the end of each subtask or as otherwise stated in the enclosure. At each of these steps Westinghouse should consult with the staff to determine if some of the calculations should be deleted or others substituted or added. Before these studies begin, Westinghouse should describe in writing how they

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propose to use these studies to resolve the issues raised by the staff. It is possible that the studies will not show the expected parametric dependence. Therefore Westinghouse should describe any modifications to the intended use of these studies after they are completed.

We would like to emphasize that the parametric approach presently being pursued by Westinghouse, while acceptable, is not the most desirable. It includes a large number of calculations that interact with one another, the review and evaluation of which places a heavy burden on our staff as well as yours and most probably does not provide justice to UHI. We strongly encourage you to generate the necessary experimental and analytical justification that would eliminate the need for the parametric approach.

Sincerely,

Original signed by
D. F. Ross

Denwood F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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Enclosure

1.0 Core Drift Flux Model

In the staff Status Report it was noted that the G-2 refill tests did not form a sufficient experimental basis for determination of suitable horizontal and vertical drift parameters. For this reason it was agreed that W undertake a parametric approach that would consider all reasonable possibilities. Three calculations were proposed (P1, P2, P3) based on a phenomenological model discussed in the SER. The parameters varied in this study were the cross-flow model and the vertical drift model in the average channel. Based on preliminary results of these calculations, it was determined that certain of the parameters proposed did not give a satisfactory description of the slip phenomena. Therefore certain modifications were agreed upon which resulted in a fourth calculation (P4). P4 is otherwise identical to P2. From these results it should be possible to determine if P4 has the least favorable combination of drift parameters, cross-flow and average channel logic. If this cannot be determined the modifications agreed upon to create P4 from P2 should be applied to the conditions of P1 and/or P3. From the results of these modified calculations the appropriate least favorable drift flux model should be chosen. Westinghouse should provide complete SATAN and LOCTA results for P1 through P4 and others if required. Included in these results should be the temperature distribution for the power regions. Westinghouse should discuss in writing the appropriateness of proposed drift flux parameters. In particular the effect of cross flow parameters, the droplet model in the average core, and vertical drift velocity, Vgj,

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should be discussed. The discussion should take account of previous calculations where support column injection location was changed and where multipliers on Vgj were used. In this regard it should be noted that Section 8.0 of this enclosure requires a sensitivity study to support column location. This is being required since it is not known how various model changes made since this selection would affect the selection.

A question concerning use of a single phase flow resistance formulation for the momentum equation in counter current flow has been raised. To assess the importance of each momentum equation term, Westinghouse should provide a term by term breakdown for one case. Westinghouse should show that the particular two-phase multipliers used in counter current flow are appropriate. Westinghouse intends to show that relatively large variations in the friction term are unimportant since its magnitude is thought to be small.

A calculation should be performed using the final drift flux model for a case using finite mixing in the upper head. This should be done on a timely basis to assure that focus on the perfect mixing case has not ignored important phenomena typical of finite mixing.

2.0 Core Flow Distribution and Unquench

The Westinghouse model, which uses uniform upper plenum and average core conditions, largely ignores enthalpy and flow distribution effects. Either experimental or analytical justification of this simple model is required. Also the status report notes that the experimental evidence has not allowed the establishment of an unquench model during counter-current flow. It has been suggested that a phenomenological model for flow distribution and unquench may not be necessary. Core flow maldistribution could result in water starvation of sections of the core. This in turn would cause these parts of the core surface to unquench. A more conservative counter-current unquench model to account for uncertainties would also cause more of the core surface to unquench. This would result in a core temperature distribution different from that predicted by the current Westinghouse model. It is this temperature distribution at the beginning of reflood which impacts the final peak cladding temperature. For this reason sensitivity studies are being proposed to explore a wide spectrum of unquench characteristics and temperature distributions. The first sensitivity study will explore temperature distribution effects. The second study will examine hydraulic feedback effects resulting from unquenching various amounts of the core at discrete times. The values of these parameters have been established as a bounds to unquench data and geometry effects. The maximum amount of core surface area assumed to be artificially unquenched in these studies at the end of active injection is assumed to be 40%. This conservatively assumes that all assemblies under guide tubes are unquenched. Under counter-current flow conditions after the upper head is empty, 2/3 of the support column assemblies are assumed to artificially unquench. This is based on a conservative overall assessment of the G-2 counter-current unquench data. This results in a total of up to 80% of the assemblies being artificially unquenched by the end of the upper head drain period.

Westinghouse intends to demonstrate that the results (peak cladding temperature) are insensitive to the parameters selected, or establish a limiting condition of these parameters for use in UHI plant calculations.

2.1 Temperature Distribution Sensitivity

A sensitivity study using LOCTA and WREFLOOD will determine the effect of core temperature distribution at the beginning of reflood (BOCREC) on peak cladding temperature. The desired result of this study would be that peak cladding temperature increases monotonically with average cladding temperature or stored energy, and is relatively independent of radial and axial temperature distribution. If this result is achieved, then the study planned in Section 2.2 can be done. If not, that study will have to be redesigned based on the results of this study. Westinghouse indicated that if the worst case is tolerable and still shows the expected behavior, they may wish to adopt that as part of their model. This would be acceptable and obviates the need for the studies described in Section 2.2. In any case, Westinghouse should justify their application of the results of this study to their UHI evaluation model.

Westinghouse attempted a similar study using the UHI evaluation model. This did not show a monotonic increase in peak cladding temperature with stored energy. This was first attributed to the conservative steam cooling model. The study was next performed using FLECHT results only, with similar results. It was suggested that the calculated time to quench the 6 foot elevation (T/TQ6) used in the FLECHT correlation was not suitable for UHI condition with large portions of cold cladding surface at the beginning of reflood. Therefore Westinghouse has proposed to use the WREFLOOD calculated T/TQ6 instead of that generated as part of the FLECHT correlation. It should be shown by comparison to applicable data that this method of calculating T/TQ6 and FLECHT heat transfer is suitable for this study.

2.1.1 The first study considers three temperature regions of the core: hot, warm, and cold. The hot region is considered to have never been quenched and is characterized by a temperature, T_{hot}, high enough to cause the maximum entrainment. The warm region is considered to be unquenched at the end of upper head draining with temperature, T_{warm}, of 500^{o}F or 600^{o}F. The cold region is the only region which remains quenched until just prior to reflood. The temperature, T_{cold}, of that region is taken to be 300^{o}F. The hot and warm regions are considered to have uniform axial unquench. That is, the entire length of the rods in these regions are considered to be unquenched at the same time. This means that the same temperature is used for T_{INIT} in the carryout correlation through out reflood. The percentages of the core which make up each region are shown in the table for the first series of calculations:

Run	% of rods at T_{hot}	% of rods at T_{warm}	% of rods at T_{cold}
1	0	0	100
2	0	40	60
3	0	80	20
4	40	0	60
5	40	40	20
6	20	0	80
7	20	40	40
8	20	60	20

The first set will be run twice with values of T_{warm} of 500°F and 600°F respectively. This series of 16 runs will be done using average mid-plane power, since the power effect on flow maldistribution will be considered in the next series and in the hot channel considerations. The hot pin temperatures, vessel inventories, and BOC REC time needed to initialize WREFLOOD and LOCTA will be from the Westinghouse "base case" used just prior to the drift flux parametric study. This model included homogenous cross flow below the transition void fraction and steam only above. The drift flux parameters are those defined in WCAP-8479-P R1.

Runs 1, 2, and 3 with both values of T_{warm} will be repeated using BOC REC initialization parameters from the finite mixing case. This finite mixing case does not have the same lateral drift parameters as the perfect mixing case. This should be satisfactory for a parametric study, but perfect mixing and finite mixing results should not be compared.

2.1.2 The analytical matrix described in Section 2.1.1 will be repeated for this series but with an additional hot region near the middle of the core. This added hot region will be the unquenched portions obtained from the power region LOCTA analysis using the hot assembly fluid conditions of the "base case". Thirty regions will be defined; 3 for each power region. The appropriate portion of the LOCTA calculated new hot region will be applied to each of the 3 regions at each power. The percentage of each of the three regions will be the same as the hot, warm, and cold regions of Section 2.1.1. The original T_{cold}, T_{warm}, T_{hot} from Section 2.1.1 will be used for calculating entrainment until the quench front reaches the LOCTA calculated hot region. Then the LOCTA calculated T_{init} will be used.

2.2 Hydraulic Feedback Sensitivity

It is expected that the studies described in Section 2.1 will determine a usable dependence of stored energy and temperature distribution on peak cladding temperature. The studies in Section 2.1 will not model any hydraulic effects that would result from artificially unquenching the various groups of rods. When rods are unquenched, the steam generation is less. Less steam generation should allow more water to penetrate the core and fall to the lower plenum, thus hastening the onset of reflood. This postulated water penetration benefit therefore has the potential of offsetting the detriment of higher temperatures at the beginning of reflood caused by unquenching. Therefore a study using SATAN as well as LOCTA and WREFLOOD will be performed in which the hydraulic effect of unquenching large sections of the core will be modeled. As before, Westinghouse should determine and justify how the results are to be used. One suggestion has been to use the worst unquench conditions as part of the evaluation model.

Four regions or channels will be defined for the SATAN analysis. Channels A, B, and C will represent large segments of the core and may be under support columns or guide tubes. Guide tube channels will have flow paths at the top of the core from the upper plenum only. Support column channels will additionally be connected to the support column volume. Table 2 defines channel arrangements for this parametric study. Support column assemblies are denoted SC & guide tube assemblies by GT. Guide tube assemblies which unquench at the end of active UHI injection are denoted by A. Support column assemblies which unquench at the end of upper head drain are denoted by U. Channels denoted by A&U will not receive water from the upper plenum after unquench. All other assemblies which are not artificially unquenched but utilize the SATAN unquench model are designated by an S. The percentage of the core occupied by A, B, and C respectively is also specified in the table. Channel D is the hot assembly. Table 2 also denotes which of the large channels provides cross flow to the hot assembly.

Run 1 is considered the base case for this study. No channels are artificially unquenched. Decreasing amounts of the core are unquenched in succession until run number in which only the guide tube assemblies are artificially unquenched. If the worst stored energy has not been reached, runs 5 and 6 will be performed in which only some of the guide tube assemblies will be unquenched at the end of active UHI injection. Additional calculations may be needed pending outcome of the results. This activity should not commence until the drift flux model has been established. Both perfect and finite mixing cases should be calculated using a $C_D = 0.6$.

Table 2

SATAN Channel Arrangements for Hydraulic Feedback Studies

	CHANNEL A			CHANNEL B			CHANNEL C			CHANNEL 4
RUN	ASSEMBLY TYPE	UNQUENCH TIME	% of CORE	ASSEMBLY TYPE	UNQUENCH TIME	% of CORE	ASSEMBLY TYPE	UNQUENCH TIME	% of CORE	CROSSFLOW CONNECTION
1	GT	S	40	-	-	-	SC	S	60	C
4	GT	A	40	-	-	-	SC	S	60	C
3	GT	A	40	SC	U	20	SC	S	40	B
2	GT	A	40	SC	U	40	SC	S	20	B
5	GT	A	20	SC	S	60	GT	S	20	B
6	GT	A	10	SC	S	60	GT	S	30	B

2.3 Heat Transfer - Flow Check

The effect of flow distribution on heat transfer within an assembly has been questioned. Westinghouse will examine the cold flow G2 tests and the uniform flow UHI heat transfer tests to see if there is any correspondence between the observed low flow areas and the hot rod behavior. Also the hot rod will be examined to see if it moves about the test assembly. Westinghouse will determine if the intra-assembly flow distributions has any effect on heat transfer. If a bias attributable to flow distribution is not discounted it must be accounted for.

3.0 Initial Upper Head Fluid Temperature

Westinghouse should provide the following:

1.) The method for calculating the initial steady-state upper head fluid temperature, guide tube and support column flows in UHI plants.
2.) A typical value of initial upper head fluid temperature.
3.) UHI set point pressure specified to preclude flashing in the upper head prior to UHI injection.
4.) Justification for the model with respect to initial temperature, uniformity and dynamics in the upper head, guide tubes, and support columns during active UHI injection.

4.0 2 Hot Channel Model

Provide results of the 2 hot channel analysis for perfect and finite mixing if 2 hot channel model is the model selected. Impact of drift flux parameters on this model should be determined.

5.0 Hot Channel Flow (Hot Rod Temperature Effects)

Determination of flow reduction (if any) to the hot assemblies should be made after items 1.0 and 2.0 have been resolved. The hot channel may be restricted from receiving water from the upper plenum and should be explored in at least one calculation that would appear to be the final model. Impact of the drift flux parameters selected in Section 1.0 should also be evaluated for this model.

6.0 High Pressure Heat Transfer

Provide the statistical analysis required to justify the high pressure heat transfer and quench models.

7.0 Final Calculation

Provide a calculation of the .6DECLG break for the perfect and finite mixing cases with all of the restrictions outlined in the status report and all modifications required as a result of these studies.

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8.0 Miscellaneous

The following items should be completed as defined in the status report:

a.) Axial shape sensitivity

b.) Psuedo viscosity study

c.) Surge tank justification

d.) Axial noding sensitivity

e.) Time step sensitivity

f.) A limited break spectrum should be calculated which includes the following:

    1.) .6 DECLG (perfect and finite mixing)

    2.) 1 $ft.^2$ break (perfect and finite mixing)

    3.) hot leg break

    4.) small break

g.) ROSA test analysis

h.) Support column location sensitivity

i.) .6 DECLG calculation repeated without UHI not functioning.