OFFICIAL USE ONLY

Docket No. 50-322

November 24, 1969

REGULATORY DOCKET FILE COPY

Report to ACRS

LONG ISLAND LIGHTING COMPANY

Shoreham Nuclear Power Station

Division of Reactor Licensing

U. S. Atomic Energy Commission

OFFICIAL USE ONLY

TABLE OF CONTENTS

ABSTRACT	Page
1.0 INTRODUCTION	1
2.0 SITE AND ENVIRONMENT	4
2.1 General Description and Population Distribution	4
2.2 Meteorology	6
2.3 Hydrology	9
2.4 Geology and Seismology	12
2.5 Foundation Engineering	13
2.6 Environmental Monitoring	14
3.0 REACTOR DESIGN	15
3.1 General	15
3.2 Core Mechanical Design	18
3.3 Reactor Control	19
3.4 Reactor Pressure Vessel	20
4.0 REACTOR COOLANT SYSTEM	22
4.1 General	22
4.2 Piping	25
4.3 Recirculation Jet Pumps	26
4.4 Main Steam Piping	27
4.5 Leak Detection	28
4.6 Inservice Inspection	29

OFFICIAL USE ONLY

Table of Contents (Cont'd)

	Page
5.0 CONTAINMENT SYSTEM	30
5.1 General	30
5.2 Functional Design of the Primary Containment	31
5.3 Structural Design of Primary Containment	41
5.4 Post-LOCA Hydrogen Control	45
5.5 Inerting of Primary Containment	46
5.6 Secondary Containment - Reactor Building	47
5.7 Reactor Building Standby Ventilating System	48
6.0 ECCS AND OTHER ENGINEERED SAFETY FEATURES	51
6.1 Emergency Core Cooling System	51
6.2 Other Engineered Safety Features	54
7.0 INSTRUMENTATION AND CONTROL SYSTEMS	55
7.1 Auto-Relief System Interlock	55
7.2 Rod Block Monitor (RBM)	56
7.3 Flow Referenced Scram	57
7.4 Common Mode Failure Study	58
7.5 Single Failure Criterion	58
8.0 ELECTRIC POWER SYSTEMS	60
8.1 Offsite Power	60
8.2 Onsite Power	62
8.3 Cable Design, Selection, Routing, and Identification	64
8.4 Environmental Testing	65

Table of Contents (Cont'd)

	Page
9.0 AUXILIARY SYSTEMS	
9.1 Shutdown Cooling System	66
9.2 Auxiliary Cooling Water Systems	66
9.3 Radioactive Waste Systems	68
9.4 Fire Protection System	70
9.5 Fuel Handling and Storage	74
10.0 STRUCTURAL DESIGN	75
11.0 ACCIDENT ANALYSIS	76
12.0 QUALITY ASSURANCE & CONTROL	79
13.0 CONFORMANCE WITH GENERAL DESIGN CRITERIA	82
14.0 TECHNICAL QUALIFICATIONS AND CONDUCT OF OPERATIONS	85
14.1 Technical Qualifications	86
14.2 Operating Organization and Training	86
14.3 Conduct of Operations	88
15.0 RESERACH AND DEVELOPEMENT PROGRAMS	89
16.0 CONCLUSION	90
APPENDIX A - Probability of an Aircraft Crash at the Shoreham Site	92
	94
APPENDIX B - Assumptions Used by the Staff in Accident Analysis	113

ABSTRACT

The Long Island Lighting Company has submitted an application for a construction permit for the Shoreham Nuclear Power Station. The nuclear steam supply system is very similar to other BWR plants which we have recently reviewed such as Hatch, Brunswick and Bell. Features unique to this facility are the geometry and design of the vapor-suppression containment and the waste gas holdup system. The initial power level of the facility is 2436 Mwt, with an anticipated ultimate power capability of 2535 Mwt.

The site on the northern shore of Long Island has a relatively low population density and satisfactory meteorology, hydrology, geology and other environmental considerations.

Subject to the resolution of a few problem areas on which we expect to be able to report at the December meeting of the ACRS, we have concluded that the Shoreham facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

1.0 INTRODUCTION

On May 15, 1968, the Long Island Lighting Company (LILCO) filed an application for a construction permit and operating license for a nuclear power plant to be called the Shoreham Nuclear Power Station. On December 3, 1968, we received a letter from the applicant informing us of their decision to defer construction and to increase the power level of the proposed Shoreham Nuclear Power Station Unit 1. Pending receipt of a revised application we forwarded to the applicant a list of questions we had on the original application and also worked with the applicant on certain siting problems and design features peculiar to the Shoreham application. On April 21, 1969, we received Amendment No. 4 to the application which consisted of a completely revised Preliminary Safety Analysis Report (PSAR) reflecting the increased power level of the plant. Amendment No. 5 to the application, dated April 25, 1969, responded to the questions we had on the original application. Table 1.0 is a list of all submittals by the applicant.

The proposed plant will be located on about four hundred fifty acres of land on the north shore of Long Island, in Suffolk County, New York. The boiling water reactor nuclear steam supply system will be furnished by General Electric Company. Stone and Webster Engineering Corporation will be the architect-engineer for the plant. Construction will be by one or more other companies, still to be selected.

The station will have an initial thermal power of 2436 Mwt, corresponding to a gross electric power output of 849 Mwe. The ultimate (stretch) power capability of the plant is anticipated to be 2535 Mwt, corresponding to a gross electric power output of 884 Mwe. The applicant's safety analysis and our evaluation are based on a plant power level of 2550 Mwt.

The design of the nuclear steam supply system is very similar to that of the Hatch and Brunswick Plants recently reviewed by the Committee. The primary containment vapor-suppression structure is unique to the Shoreham Plant. It is a steel-lined, reinforced concrete structure with a cylindrical lower section and an upper section in the shape of a conical frustrum, within which the drywell and wetwell are separated by a concrete floor. The radioactive waste gas system is also different in that it incorporates several large decay tanks which provide a longer holdup period for waste gases than is generally provided in BWR plants, and in that there is no plant stack. Waste gases will be released from a vent on the reactor building roof.

TABLE 1.0 LIST OF SUBMITTALS BY THE APPLICANT

Amendment	Date	Subject
———	May 15, 1968	Initial application and PSAR Filed
No. 1	June 18, 1968	Clarification of earliest and latest completion dates
No. 2	September 26, 1968	Revision and additions to PSAR
No. 3	February 5, 1969	Supplementary information pertaining to proximity of airport to the Shoreham site
No. 4	April 21, 1969	Completely revised PSAR reflecting increase in proposed power level of plant
No. 5	April 25, 1969	Response to DRL request of January 21, 1969, for additional information
No. 6	July 1, 1969	Corrections and revision to Amendments 4 and 5
No. 7	August 27, 1969	Outstanding information from January 21, 1969, DRL request, and response to subsequent oral request for additional information
No. 8	October 24, 1969	Revised and supplementary information
———	October 24, 1969	Proprietary submittal on LOCTVS computer code
No. 9	November 19, 1969	Revised and supplementary information on unresolved items

2.0 SITE AND ENVIRONMENT

2.1 General Description and Population Distribution

The proposed Shoreham Nuclear Power Station is situated on the north shore of Long Island in the town of Brookhaven, Suffolk County, New York, approximately 45 miles east of New York City. The site is a 450 acre tract of land owned by LILCO. The site property is wooded and hilly, rising from zero feet MSL at the shoreline to 40 feet MSL in the reactor building, to 150 feet at the highest point on site. The nearest residence is approximately 1500 feet from the reactor building. The nearest property boundary is approximately 1000 feet from the reactor which also is the minimum exclusion zone radius.

The land area within five miles of the site is relatively sparsely populated (1960 population 7500) and the land within this area is largely reserved for special, nonresidential, long term purposes, i.e., RCA station near Rock Point, Brookhaven National Laboratory (BNL), Grumman Peconic River Airport and the Wildwood State Park. It is anticipated that the population density within this area will continue to be low (about 190 people/mile^2).

The applicant has stated, and we agree, that a five mile low population zone radius is available at this site. We have reviewed the applicant's analyses and have determined that the Part 100 guidelines for this site, with respect to the available exclusion and low population zone distances (1000 feet and 5 miles, respectively), can be satisfied.

As indicated in our reports to the Committee of October 30, 1968 and March 20, 1969, the proposed Shoreham Plant will be located about 4-3/4 miles from the Grumman Aircraft Company (Peconic River) Airport. We and the applicant (ref. Comment #2.1, Amendment 5) have investigated the probability of an aircraft from this airport crashing into the proposed facility.

We determined the types of aircraft using the Grumman Airport and obtained crash statistics for these various types of aircraft. Using these data, we determined the relative probability of an aircraft crash as a function of distance from the airport. We examined the effect of using only fatal crash statistics as opposed to total crash statistics, the effect of using different analytical techniques in our calculation (e.g. various geometrical flight paths), and the effect of the several different types of aircraft using the Grumman Airport on the probability of crashes at the Shoreham site. The details of our analysis are presented in Appendix A to this report. Based upon our analysis, we conclude that the site is sufficiently distant from the Grumman Airport that the proposed Shoreham Plant need not be designed with special provisions to protect the facility against the effects of an aircraft crash.

2.2 Meteorology

Because of the coastal location, the site is subject to offshore winds at night and onshore winds during the day. This diurnal variation in addition to rather frequent frontal passages, tends to reduce the probability of winds persisting in any wind direction for a prolonged period of time. The average wind velocities at the site also tend to be higher than at most inland locations. The site has somewhat better potential atmospheric dilution than the average site. The discussion of the proposed hurricane protection for the facility is presented in the following section on hydrology.

The meteorological diffusion parameters used in the applicant's accident analyses are based upon ten years of meteorological data collected at the Brookhaven National Laboratory (BNL) located six miles to the south of the site. Although the topography of BNL and the Shoreham site are somewhat different and the site is closer to the water, we and our consultants agree with the applicant that the BNL data provide a reasonable estimate of the expected meteorological diffusion conditions at the site. The applicant initiated an onsite meteorological program in September 1967 which includes the measurement of wind speed, vertical and azimuthal wind direction and temperature lapse rate with height, as measured on a 135-foot tower at an inland location on the site; and wind speed and azimuthal wind direction, measured on a short pole in the beach area. Also temperature and precipitation are measured in the beach area. The data being collected will provide a basis for estimating the degree of conservatism of the accident meteorology and provide an adequate basis upon which the routine gaseous release limit will be set at the operating license stage of review for this facility.

Results from wind tunnel studies conducted by the applicant (ref. Comment #9.7, Amendment 5) show that for stack flows corresponding to the accident mode of operation of the standby ventilation system, a release from a vent on the reactor building roof may rapidly be brought down to the ground by aerodynamic downwash in the wake of the building. For this reason, both we and the applicant have assumed a ground release for our accident dose estimates.

BNL has developed a system of diffusion model categorization, (similar to but different from the Pasquill categorization) for the area based on ten years of meteorology data. Since the BNL parameters were derived specifically for the general area of the site we have concluded that they are more appropriate for this site than are the Pasquill parameters. Therefore, both we and the applicant used the BNL categorizations and parameters in calculating potential offsite doses which might result from postulated accidents. The BNL parameters (moderately stable condition) used for the 0 to 24 hour period are slightly more conservative than the Pasquill Type F which is normally used.

The meteorological model used by us in calculating potential accident doses is as follows:

0-12 hours	- BNL Type D, 1 m/sec, building wake and invariant wind direction
12-24 hours	- BNL Type D, 2 m/sec, building wake and invariant wind direction
1-4 days	- BNL Type C, 4 m/sec, and uniform mixing into a 22-1/2 degree sector
4-30 days	- 50% BNL Type D, 2 m/sec, 50% BNL Type B, 4 m/sec, and 25% frequency in a 22-1/2 degree sector

The applicant used the same meteorological assumptions for the 0 to 24 hour period. However, he used somewhat less conservative parameters for the one to thirty day period than we used.

Our meteorological consultants from ESSA, whose report was previously sent to the Committee, agree with our conclusions that the BNL parameters can be used for this site, that the 0 to 24 hour meteorology parameters proposed by the applicant are adequately conservative, and that the one to thirty day meteorology parameters used by the applicant are not adequately conservative. For the latter case, the applicant selected parameters based primarily on data pertaining to steadiness of wind direction while ignoring associated factors such as wind speed and inversion frequency. Our review of the meteorological data shows that thiry day periods of less favorable dispersion characteristics than characterized by the applicant actually exist, and our model has been developed taking this into account. A more conservative model suggested by ESSA for the thirty day period, while not identical to ours, yields essentially the same calculated doses.

2.3 Hydrology

Cooling water for the proposed Shoreham facility will be taken from Long Island Sound and be returned to the Sound along with periodic additions of liquid radioactive wastes. Circulating water discharge is from a multi-port diffuser located 1600 feet off shore. The diffuser consists of a number of submerged jets which propel water horizontally from the jets at initial velocities of the order of 8 fps. This jet action will provide some additional dilution of the discharge water.

The applicant has provided a reasonable estimate of the dilution of liquid effluents in the Sound. Some of the effluent which is moved out from the site area during ebb tide may be returned with the flood tide. However, a buildup of effluent does not appear to be a problem since there are no bays or inlets in the area to trap effluents. The net transport out of the Sound is of the order of 50,000 cfs. There do not appear to be any hydrologic conditions which could present a problem relative to the routine release of liquid effluents in compliance with the 10 CFR 20 limits.

All public and domestic water supplies in Long Island are derived from the ground water. Any spill of radioactive liquids onto the ground will run off directly into the Sound. If the spill reaches the ground water it would flow directly or indirectly into the Sound via the streams surrounding the site. Ground water flow would be from the site toward the drinking water supplies only if the present ground water gradients were reversed, a condition which would also lead to salt water intrusion. Since salt water intrusion would ruin the drinking water, New York State policy for the control of ground water use will preclude this situation from developing.

The comments of our hydrologic consultants at the USGS will be forwarded to the Committee prior to the December meeting. The USGS has told us informally that the applicant's comments concerning estimates of the dilution of effluents in Long Island Sound and the ground water hydrology are reasonable.

The applicant has estimated the peak storm surge at the site that could result from the occurrence of the probable maximum hurricane (PMH). The hurricane parameters used in calculating the hurricane surge were those defined in ESSA report HUR 7-97 "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coast of the United States". The storm was superimposed on a spring high tide, and the resulting still water level at the Shoreham site was calculated to be 15.8 feet above the Mean Low Water (MLW) level. It was estimated that the peak level, including the runup of waves at the site, would be 20.5 feet above MLW. Station grade will be 20 feet above MLW. Based upon the PMH estimates stated above, the applicant proposes to protect all components necessary to maintain the station in a safe shutdown condition against inundation and wave runup to 25 feet above MLW.

The minimum low water level for the site was calculated in a manner similar to the high water level and an estimate of 4.7 feet below MLW was made by the applicant. Water will be supplied to the station through the intake canal, the bottom of which is at 12.0 feet below MLW and which terminates at the screen well at 21.0 feet below MLW. The service water pumps take suction from the screen well.

Our consultants at the Coastal Engineering Research Center have noted that the applicant did not consider the possiblity of forerunner surge, bathostrophic tide and a high spring tide in estimating the peak storm surge at the site. We have discussed this with the applicant and he has agreed to revise his analysis to consider these phenomena. This analysis cannot be completed in time for the ACRS meeting. The applicant has stated, however, that he will design the plant to protect vital structures and components against the peak storm surge, including runup of waves, that is determined by an analysis acceptable to us and our consultants.

2.4 Geology and Seismology

The site is located in the Atlantic Coastal Plain Geologic Province. Details of the geologic structure in the crystalline basement rocks, which are overlain by more than 1300 feet of consolidated and unconsolidated sediments, are not well known. The crystalline rocks are of Paleozoic age and are similar to the basement rock in the Piedmont Geologic Province to the west.

The Connecticut Valley fault forms the eastern border of the Connecticut Valley Lowlands which is the geomorphic expression of a Triassic Basin north of the site. The seismic activity in central Connecticut is associated with this structure. Indications are that this fault may continue south into Long Island Sound and possibly to within approximately 10 miles of the site. No other major geologic faults are known that could localize seismicity near the site.

The geological and seismological characteristics of the site area require the assumption that earthquakes with bedrock intensities characteristic of the Piedmont Province and surface intensities characteristics of the Coastal Plain Province might occur near the site. The upper strata of unconsolidated sediments overlying the site are loose to medium density and therefore could amplify any bedrock vibrations that might occur.

Based on the review of the earthquake activity in Connecticut, New Jersey, and New York and of foundation conditions at the site, our consultants recommend horizontal seismic design accelerations of 0.10g and 0.20g for the Operating Basis Earthquake and Design Basis Earthquake, respectively. The vertical design accelerations should be at least two thirds those of the horizontal design accelerations. The applicant has agreed to use these values in the design of the facility.

2.5 Foundation Engineering

Approximately 1100 feet of unconsolidated sediment underlie the proposed plant site. The shallow, unconsolidated strata which will support the plant structures consist of loose to dense sands. The applicant will excavate the uppermost strata under all principal structures down to elevation -12 feet MLW and replace it with compacted fill. The reactor containment building, the deepest structure, will be founded at elevation -2 feet.

The applicant analyzed the sand strata below elevation -12 feet for stability under dynamic stress (seismic) conditions. The resulting factors of safety against liquefaction were lowest at elevation -40 feet, ranging from 1.6 at the intake structure to 2.15 at the reactor containment building. However, in computing the shear stresses in the sand, the applicant reduced by approximately one third the 0.2g acceleration which we and our consultants recommended for the DBE. The rationale for this reduction in seismic acceleration was not acceptable to us or our seismic consultant, Newmark and Associates. Our seismic consultant therefore made an independent analysis. Although lower factors of safety were computed by our consultant, he believes the foundation soils will be stable under the dynamic stresses from a 0.2g earthquake. The least stable condition for the soil strata under the plant was determined to occur during the DBE. Foundation conditions appear to be adequate for all other load conditions.

2.6 Environmental Monitoring

Environmental radiation monitoring data are available in the general site area for approximately 20 years. These data were collected in connection with operations at BNL. Background radiation levels in milk, water, vegetation and fallout have been well established.

The applicant proposes to initiate an independent preoperational environmental radiation monitoring program approximately two years prior to plant operation. The applicant proposes to collect samples of air, surface water, bottom sediment, aquatic biota, soil, milk, food crops, and vegetation. The applicant will also conduct a marine ecological program which will include a study of water temperatures, salinity, bottom composition, water chemistry, bottom biota, plankton, crustacea and fish to determine background aquatic conditions prior to plant operation. Some of this work has already been initiated. The environmental monitoring and ecological studies proposed by the applicant should provide a sound base upon which to develop operational programs to determine the effects of plant operation on the environment.

3.0 REACTOR DESIGN

3.1 General

The reactor design of the Shoreham Plant is similar to that of several previously reviewed plants. Table 3.1 provides a comparison of the reactor design parameters for the Shoreham, Brunswick, Hatch, and the Cooper facilities.

It is evident from these tabular data that the Shoreham reactor is of the same class of reactors as those used for the Brunswick, Hatch, and Cooper Plants with respect to thermal and hydraulic parameters. The only significant difference is the lower (2.00 w/o vs 2.15 to 2.25 w/o) average initial fuel enrichment. There is a corresponding reduction in the average exposure of the fuel at discharge, (16,680 vs 19,000 MWD/MTU). The changes are apparently due to a revised economic optimization.

TABLE 3.1

COMPARISON OF REACTOR DESIGN PARAMETERS

PARAMETER	SHOREHAM	BRUNSWICK	HATCH	COOPER
	50-322	50-324/325	50-319	50-298
Rated Core Thermal Power (Mwt)	2436	2436	2436	2381
Net Electrical Output (Mwe)	819	821	786	778
Maximum Specific Power (kw/ft)	18.3	18.5	18.3	18.5
Average Specific Power (kw/ft)	7.1	7.1	7.1	7.1
Maximum Heat Flux ($BTU/hr\text{-}ft^2$)	428,308	428,967	428,300	427,820
Average Heat Flux ($BTU/hr\text{-}ft^2$)	164,734	164,743	164,734	164,500
Feedwater Temperature (F)	420	420	387	367
Core Average Exit Quality (%)	14.1	14.1	13.9	13.4
Initial Fuel Enrichment,				
Averaged Over Each Assembly, (w/o)	2.00	2.25	2.23	2.15
Average Discharge Exposure (MWD/MTU)	16,680	19,000	19,000	19,000
Moderator Temperature Coefficients				
Cold, ($\Delta k/k/°F$)	-5.0×10^{-5}	-5.0×10^{-5}	-5.0×10^{-5}	-5.0×10^{-5}
Hot (No Voids), ($\Delta k/k/°F$)	-3.9×10^{-4}	-1.7×10^{-4}	-3.9×10^{-4}	-3.9×10^{-4}

OFFICIAL USE ONLY

TABLE 3.1 (CONT'D)

PARAMETER	SHOREHAM	BRUNSWICK	HATCH	COOPER
Equivalent Core Diameter, (in.)	160.2	160.2	160.2	160.2
Number of Fuel Assemblies	560	560	560	560
Fuel Rod O.D. (in.)	0.563	0.563	0.562	0.562
Fuel Pellet Diameter, (in.)	0.487	0.487	0.488	0.488
Clad Thickness, (in.)	0.032	0.032	0.032	0.032
Uranium Weight per Assembly, (lb)	429.66	429.66	429.66	429.66
Reactor Vessel I.D., (in.)	218	218	218	218
Reactor Vessel Wall Thickness (in.)	5-17/32	5-17/32	5-17/32	5-17/32

OFFICIAL USE ONLY

3.2 Core Mechanical Design

For normal design loads of mechanical, hydraulic, and thermal origin, plus the loads resulting from the operational basis earthquake, the reactor internals will be designed to function within the stress limit criteria of Article 4, Section III of the ASME Boiler and Pressure Vessel Code. Where deflections must be considered in component design, the deformation limits of the nuclear system loading criteria, discussed in Section 4.0, will apply. Under the above loading conditions, these criteria require that deflections be limited to less than half of those calculated to cause loss of function. These design limits are acceptable.

Under hypothetical accident conditions, which include the combined loads from a recirculation line break or a steam line break plus the design basis earthquake for the Shoreham plant, the primary design objectives for the reactor internal structures require that the core reflooding and cooling capabilities be maintained, that no item which could block the main steam line isolation valves will fail in such a manner as to be discharged through the main steam line, and that the control rods will operate. The stress limits for these conditions are those of the nuclear steam system loading criteria. For the loading combinations of normal plus the DBE or normal plus pipe rupture, which bound these accident conditions, the deflections allowed by these loading criteria will be limited to less than 2/3 of those causing loss of function.

Our review of these loading criteria has shown that the margins of safety provided are essentially those which have been previously accepted. We therefore, conclude that these design objectives and limits are acceptable.

3.3 Reactor Control

Reactor control is accomplished by the use of 137 cruciform control rods actuated by hydraulic drive mechanisms which are identical to those in other recent boiling water reactors.

In addition to the control rods, there is a standby liquid control system which can inject sodium pentaborate to provide an independent way of shutting down the reactor. As in other BWR plants, this system does not provide a rapid scram function and does not alleviate the consequences of a design basis accident. It is therefore not considered to be an engineered safety feature and does not meet all the usual requirements of redundancy, capability to accommodate single failures and the IEEE criteria for associated instrumentation and controls. As on previous plants, we have concluded that this is acceptable.

3.4 Reactor Pressure Vessel

The applicant is procuring the reactor pressure vessel originally intended for the N. Y. State Electric and Gas Company Bell Station for use in the Shoreham Station. The vessel, which is partially fabricated (approximately 28% complete), was ordered from Combustion Engineering by GE on February 1, 1967. The initial specification required conformance to the 1965 edition of Section III of the Code, including the Winter 1966 Addenda. Additional requirements will be imposed on the remaining fabrication work in order that the vessel, as finally installed in the Shoreham Plant, will meet the intent of the 1968 edition of the code to the maximum extent possible. The applicant investigated what will be involved in this additional effort and concluded that, because GE and CE routinely specify nondestructive testing which exceeds Code requirements (e.g., 100% volumetric inspection of the vessel), only a relatively few additional requirements would need to be imposed. Certain documentation records which were already completed on this vessel are not consistent with present Code requirements. For this reason the vessel can not be identified as a 1968 Code vessel and will be stamped in accordance with the 1965 edition of ASME Code, Section III.

We have reviewed the information submitted by the applicant, including all the requirements to be applied to the vessel in addition to those in the 1965 edition of the ASME Code, Section III. Our review encompassed the quality control provisions, nondestructive examination procedures, vessel fabrication, fabrication and material identification records, inspector qualification, and vessel design. We have concluded that the Shoreham vessel is acceptable.

The applicant estimates that the end-of-life fluence to the reactor vessel is 7.2×10^{17} nvt. The surveillance program provides three specimen baskets to be placed in the reactor initially and a fourth basket to be held in reserve for contingencies. Our independent estimate of the end-of-life fluence indicates that it may be as high as 1.8×10^{18} nvt. We therefore intend to require the applicant to attach a capsule containing dosimetry wires to one of the three baskets in the vessel. This dosimetry capsule will be withdrawn at the first refueling to verify the perdicted fluence. If extrapolation of the dosimetry measurements indicates that the total vessel exposure may exceed 10^{18} nvt, the applicant will be required to install the fourth specimen basket. These provisions assure that the Shoreham materials surveillance program will be sufficiently flexible to meet our requirements.

4.0 REACTOR COOLANT SYSTEM

4.1 General

The principal design parameters of the reactor coolant system are shown in Table 4-1.

Table 4-1

Design Thermal Power	2436 Mwt ($8.33 \times 10^9 \frac{BTU}{hr}$)
Design Pressure (psig)	1250
Design Temperature	575°F
Total Core Coolant Flow Rate (full power)	75.5×10^6 lb/hr
Steam Flow Rate (full power)	10.47×10^6 lb/hr
Normal Operating Pressure (psig)	1005

The stress, deformation, fatigue and buckling limits originally proposed in Appendix D of the Shoreham PSAR were similar to those which we did not find acceptable during the Brunswick and Hatch reviews. These limits were modified (Amendment 7) and now are the same as those agreed upon for the Brunswick and Hatch plants. In addition to increasing the factors of safety to levels more consistent with the intent of the various codes, these amendments offer a commitment to discuss, before use, the details of proposed empirical techniques which may be used to reduce the safety margins provided in the design of critical structures for hypothetical accident conditions. The use of these new limits gives essentially the same margins of safety relative to stress and fatigue that would result from the use of applicable portions of Section III of the ASME Boiler and Pressure Vessel Code and the

B31.1.0, and B31.7 Piping codes. We therefore, find the nuclear steam loading criteria for the Shoreham Plant acceptable. The NDTT criteria for the reactor coolant system given in the Shoreham PSAR (as modified by Amendment 7)* are essentially a reiteration of the statement of implementation of General Design Criterion 35, which we developed during the Hatch and Brunswick reviews and which was incorporated in the commitments made for these plants. We therefore find the Shoreham proposal acceptable.

Potential vibration loads in the nuclear steam supply system will be considered as a part of the mechanical design criteria. Quantitative limits on amplitude and frequency have not been included in this application. The applicant has, however, stated that the general stress, deflection and fatigue limits given for the nuclear steam system will apply. We consider this acceptable at the construction permit stage.

---

* Statement of implementation of General Design Criteria 35

a. Piping and pressure containing parts with a wall thickness greater than 1/2 in. will have a nil ductility transition temperature, by test, 60°F below anticipated minimum operating temperature when the system has a potential for being pressurized to above 20 percent of the reactor design pressure.

b. Those pipes and pressure containing parts with a wall thickness 1/2 in. or less need not have material property tests (such as the Charpy V-notch) if:
   1. They are fabricated from austenitic stainless steel
   2. The material has been normalized (heat-treated)
   3. The material has been fabricated to "finegrain practice"

Extensive vibration test programs are now being conducted at several large BWR plants. Aside from verifying the adequacy of the vibration control programs for the present operational requirements of these particular plants, the data from these tests will constitute a significant contribution to the information necessary to quantitatively evaluate the long term performance of the BWR reactor system.

The applicant has made no commitment to perform vibration testing. The Nuclear System Supplier, General Electric Company, has stated in meetings on Shoreham and other BWR plants that the results of tests on prior plants of similar design will be adequate to assure the performance of the later plants, and that, in any event, nothing will be done to preclude vibration testing at the operating stage if it should be shown to be necessary. Both the applicant and General Electric Company are aware of this position and of the Staff and Committee's desire to have serious consideration given to in-service monitoring of vibration.

We intend to require the performance of confirmatory testing at the operating license stage.

4.2 Reactor Coolant Piping

The reactor coolant piping and valves will be designed to the USAS Code for Pressure Piping B31.1.0-1967 plus a number of additional requirements. These requirements place limitations on the materials which may be used and define the nondestructive testing to be conducted.

The material specifications cited in these requirements allow only seamless pipe or welded pipe formed of high quality plate with all seam welds examined by dye penetrant or magnetic particle methods in accordance with Section III of the ASME Boiler Pressure Vessel Code and by radiography in accordance with applicable ASTM Specifications.

In addition to material limitations, the additional requirements include: full radiographic examination of all girth welds in piping over 2 inches in diameter and of all branch connecting welds over 4 inches in diameter; the exclusion of backing rings from all welded joints; and surface examination of all girth welds regardless of size. This proposed test program upgrades the nondestructive testing requirements to essentially those of the B31.7 Code for Nuclear Power Piping.

We find the design criteria, materials limitations, and the proposed testing program for the reactor coolant system acceptable.

As in the case of the majority of BWR plants now in operation or being designed, the recirculation pumps will be designed to Section III of the ASME Boiler and Pressure Vessel Code as Class C vessels. We find these design criteria to be acceptable.

4.3 Recirculation Jet Pumps

Each recirculation loop will connect to ten jet pumps within the reactor vessel. The jet pumps, which have no moving parts and operate on the principle of converting momentum to pressure, are identical to those used in other recent GE BWR's.

The mechanical design criteria for the jet pump assemblies are the same as those for the rest of the core internal structures discussed in Section 3.2 above. The jet pumps will receive the most severe loading under the conditions which would result from a loss-of-coolant accident and subsequent operation of the emergency core cooling systems. The primary stresses under these conditions are within the limits of the nuclear steam system loading criteria.

We have concluded that the proposed design criteria and preliminary designs of the jet pumps are acceptable.

4.4 Main Steam Piping

The main steam piping and the branch lines from this piping will be designed to the USAS B31.1.0-1967 Code for Pressure Piping. We find the stress limits and design techniques thus defined to be adequate. The applicant has agreed to do a modified dynamic analysis of the main steam piping between the second isolation valve and the turbine. This analysis will include branch lines (larger than 2-1/2 inches diameter) up to and including the first isolation valve on each branch.

All welds in the main steam piping from the reactor vessel to the anchor point downstream of the second isolation valve will receive 100 percent radiographic examination. From this point, up to but not including the turbine stop valve, the applicant has proposed to perform only spot radiography (20%) of all welds. It is also proposed that no inspection requirements other than those of the B31.1.0-1967 piping code be placed on the branch lines. Since the code nondestructive testing requirements are optional in the case of all these lines (because of the system and/or wall thickness) there is essentially no inspection commitment made for these branch lines.

We do not find this approach acceptable. We plan to require that all pipe welds in the main steam lines from the reactor vessel to turbine and in branch lines (over 2-1/2 inches in diameter) up to and including, the weld to the first isolating valve on each branch line receive 100% volumetric examination, as well as surface examination by either liquid dye penetrant or magnetic particle techniques. The applicant has stated orally that he may accept this degree of inspection but has not yet made a commitment.

The proposed design criteria for the main steam lines include no explicit provision for inspection of valves which form a part of the pressure boundary. The principal argument offered is that standard valves downstream of the main steam isolation valves would not pass the usual acceptance standards for radiographic inspection and yet have an excellent service record in conventional plants. We do not consider this to be sufficient justification for leaving the quality of such components in question. We plan to require that all cast fittings and all pressure boundary parts of valves over 2 1/2 inches in size and associated welds receive 100% volumetric and surface examination.

4.5 Leak Detection

The leakage limits proposed in Amendment 8 of the PSAR are 15 gpm for unidentified and 50 gpm for total leakage, (identified plus unidentified). We have informed the applicant that we will require a 5 gpm limit for unidentified leakage and a 25 gpm limit for identified leakage.

The applicant has proposed monitoring leakage flows to an equipment drain sump and to a floor drain sump. Periodic pump-down of drain sumps has proven to be a reliable and sensitive but a slow means of detecting leakage from the primary system. In addition he states other methods of primary coolant leak detection will be considered for Shoreham. We expect to require that at least one additional system, specifically designed to rapidly detect primary coolant leakage, be employed at Shoreham. We will continue our review of the development of an acceptable redundant leak detection system during the construction phase.

4.6 Inservice Inspection

The applicant has not yet submitted an inservice inspection program for our review. He has told us that he has initiated a detailed study to establish a comprehensive inservice inspection program for the Shoreham Plant. The ASME "Code for Inservice Inspection of Nuclear Reactor Coolant System" will be used as a guide in developing this program and to design systems so as to provide adequate access. He will submit an interim status report on the development of this program in about six months and a final report by January 1, 1971.

We have informed the applicant that we will require him to conduct a base line inspection after the reactor primary coolant system hydrotest and prior to startup. The program will also include the engineered safety features and the main steam lines between the second isolation valve and the turbine stop valves.

We have concluded that these provisions for inservice inspection are satisfactory for a construction permit.

5.0 CONTAINMENT SYSTEM

5.1 General

The design of the Shoreham containment system is similar to that of other recent BWR facilities in that it has a vapor-suppression primary containment within a secondary containment building. The primary containment is different from the usual light bulb and torus steel vessels in that it is a steel-lined reinforced concrete structure. The geometry of this structure is unique, consisting of a conical frustrum over a cylindrical section, with the drywell in the upper conical section and the wetwell or suppression chamber in the lower cylindrical section. The function of the Shoreham primary containment, as is that of its steel counterpart in other BWR facilities, is to absorb the energy release from a loss-of-coolant accident (LOCA) and provide a low leakage barrier to the release of fission products. The design leakage rate for the Shoreham primary containment is the same as that for most steel vapor-suppression primary containments - 0.5% per day at design pressure.

The secondary containment or reactor building will be designed to have limited leakage, as discussed in greater detail in Section 5.5 below. During normal operation, the building's regular ventilation system will maintain it at a slightly negative pressure (about 1 in. of water), so that all leakage will be into the building. The filtered discharge is from a vent on the roof.

In the event of an accident, the normal ventilating system for the reactor building would be automatically shut down and the standby ventilation system actuated. This system is similar to that provided in other BWR vapor-suppression facilities which we have reviewed in that one of its functions is to provide absolute and charcoal filtering of discharge air during an accident. In the Shoreham Plant, however, this system provides another function, i.e. to eliminate the possiblity that fission products which leak from the primary containment could pass directly into the discharge stream to the filters without first being mixed with the reactor building air (see Section 5.6 below). This secondary function is necessary in the Shoreham Plant in order to make acceptable the calculated potential offsite doses from the design basis accidents, because this plant, unlike most other BWR plants, has no stack to provide the additional dilution associated with an elevated release.

5.2 Functional Design of the Primary Containment

The analysis of LOCA pressure and temperature transients in the Shoreham vapor-suppression containment was performed by the architect-engineer, Stone and Webster, using their own proprietary computer code, LOCTVS (Loss of Coolant Transient Vapor-Suppression). Previously, General Electric has always calculated the pressure transients for BWR vapor-suppression containments. The primary containment design pressure for Shoreham is lower than that for all previous BWR plants.

A major new consideration associated with the Shoreham primary containment design is the essential integrity of the drywell floor or deck during a LOCA. During the early stages of LOCA blowdown, the pressure begins to rise in the drywell and a force is exerted on the deck and on the water that fills the bottom eleven feet of the vent pipes. This force accelerates water in the vent pipes down into the wetwell pool. Once the pipes are cleared of water, a mixture of steam, water, and air flows from the drywell into the wetwell pool, wherein the steam condenses. However, until the vent pipes are cleared, the drywell pressure rises very rapidly at (about 40 psi/sec) for about half a second, at which time the vents are cleared. During this period a differential pressure will exist across the deck, peaking at about 20-25 psig. The deck must withstand this peak differential pressure for the pressure suppression system to perform properly. If it does not, the design pressure of the primary containment may be considerably exceeded.

Special attention must also be given to potential flow paths that could connect the drywell directly to the wetwell air volume during a LOCA. A feature of the Shoreham containment is a flexible seal between the deck and the walls of the primary containment. This seal must be capable of withstanding the combined effects of blowdown jet forces, the temperature and pressure transient associated with a loss-of-coolant accident, seismic events that may give differential motion to the deck and the primary containment liner, and differential thermal expansion. The applicant has stated that this seal will be designed for zero leakage.

A similar concern exists with the vacuum breakers between the drywell and wetwell. These vacuum breakers permit the return of air from the wetwell to the drywell when the drywell pressure decreases below the wetwell pressure by some small amount (usually 0.5 psi). The vacuum breaker system is a potential leakage path between the drywell and wetwell air volume. The applicant has stated that the final design of the Shoreham primary containment will be such that any one vacuum breaker valve could be fully open during a design basis LOCA and not exceed the design conditions of the structure. He has indicated that this will probably be accomplished by having two vacuum breaker valves in series or by having small enough valves taht the bypass flow though one valve could be accommodated.

An important consideration in the Shoreham containment is the difference in downcomer vent system design from that used in previous BWR designs. The Shoreham vent configuration is geometrically simpler than that used in other BWR's. In some BWR's, the large vent pipes, often 6 to 8 feet in diameter, are connected to a ring header from which about 96 downcomer pipes lead into the pool water with a submergence of about 3 to 4 feet. The Shoreham vent system design consists of seventy-four, 47-foot-long straight pipes, each with a submergence of eleven feet. This design results in a vent loss coefficient of 2.1, compared with loss coefficients of about 6.2 for the ring header configuration. The lower coefficient results in a lower peak drywell pressure, and a more rapid energy addition to the pool water during blowdown. As part of our review, we assured ourselves that the energy deposition rate to the pool was low enough to allow complete condensation of the steam carried through the vent system.

5.2.1 Calculation Method

Our detailed review of the calculational method used by Stone and Webster leads us to conclude that the LOCTVS code conservatively calculates the peak drywell, wetwell, and deck differential pressures for the Shoreham primary containment. This conclusion is based on the following:

1. LOCTVS has been used to calculate the pressure transients observed in many of the Moss Landing tests. In all cases, LOCTVS calculated peak pressures are equal to or higher than those observed in the tests. A major reason for the LOCTVS overprediction is its use of the Moody blowdown model which adds mass and energy into the drywell more rapidly than it could actually happen for the larger, critical breaks.

2. The peak pressure is calculated neglecting steam condensation on drywell structures and with the assumption that 100% of the water added to the drywell during blowdown is carried into the suppression pool. These assumptions are conservative and account for the major effects of containment prepurging.

3. The pressure transient is calculated assuming initial conditions which maximize the containment pressure. These initial conditions include the assumption of instantaneous closure of the steam isolation valve, no feedwater flow, a high air mass inventory in the drywell and a high initial containment pressure.

4. The peak differential pressure across the deck has been conservatively calculated. LOCTVS consistently calculates a vent clearing time that is longer than those observed in the pressure suppression tests. Since the deck differential pressure continues to rise until the vents are cleared, LOCTVS overpredicts the deck differential pressure. Use of the Moody blowdown model also adds to the conservatism of the deck differential pressure.

5. We have compared the Shoreham pressure transients calculated by LOCTVS with those calculated independently by us using the CON-PS code, and the agreement is excellent. CON-PS is a pressure suppression code that has been developed by the Idaho Nuclear Corporation for us in the Division of Reactor Licensing Technical Assistance program.

5.2.2 Comparison of Shoreham Design with Experimental Configurations

The satisfactory performance of pressure suppression containment depends on complete condensation of the steam that is transported through the vent system into the suppression pool. Incomplete condensation will result in high pressure in both the drywell and the wetwell. A complete understanding of how all the parameters which may affect complete condensation has not been obtained. While modest extrapolations of some system parameters may not significantly affect the predictions of complete steam condensation, the present state of technology makes large extrapolations inadvisable. As shown in Table 5.2 and discussed below, the critical parameters of the Shoreham design are all either within the ranges of the parameters used in the Moss Landing tests, or in the obviously safe direction.

Local overheating of the pool water, sufficient to prevent total condensation, is possible if the diameter of the downcomer pipe is too large, of if the downcomer pipes are spaced too closely together. The downcomer pipes proposed for the Shoreham have the same diameter as those used in the Moss Landing tests and a favorably greater center-to-center spacing. Placing the downcomer pipes too close to the pool bottom could also prevent complete condensation by allowing the steam-air jet to be reflected off the bottom and be redirected towards the pool surface. Based on test data, the Shoreham design has an adequate eight foot clearance between the bottom of the pool and the downcomer pipes. If the pool depth is not sufficient, some bubbles could pass through the pool water without condensing. The depth of the Shoreham pool is 18 to 19 feet, which is considerably greater than that in other BWR designs.

TABLE 5.2

COMPARISON OF PARAMETERS FOR TESTS AND FOR SHOREHAM DESIGN

Parameter	Moss Landing Test	Shoreham Design
Downcomer Pipe Diameter, in.	14 to 24	23.5
Downcomer Spacing, ℄ to ℄ ft	3.67	6
Submergence, ft	-2 to +12.5	11
Downcomer Distance to Pool Bottom, ft	6 to 12	8
Breaker Area/Vent Area Ratio	0.0015 to 0.0485	0.019
Vent Loss Coefficient	5.6	2.1
$\frac{\text{Pool Surface Area, ft}^2}{\text{Vent Pipe Area, ft}^2}$	$\frac{21.5}{1.07}$ (Humboldt)	$\frac{61}{3.1}$
	$\frac{44}{3.1}$ (Bodega)	
Maximum Pool Temperature during Blowdown, °F	163	149

The only significant deviation of the Shoreham design from the Moss Landing test conditions is the geometry of the vent system. The Shoreham design has a vent loss coefficient of 2.1, consisting of the frictional loss through the 48-foot straight pipe, plus inlet and exit losses. The Moss Landing test configurations and previous light bulb and torus containment designs had a calculated vent loss coefficient of 5.6, consisting of losses in the inlet and outlet, two tees, one elbow, and 45 feet of pipe. The most important effect of a lower vent loss coefficient is to reduce the drywell peak pressure. In previous vapor-suppression containment designs (by GE) 10% was added to the calculated vent loss coefficient of 5.6, resulting in a vent loss coefficient of 6.2 for design purposes. If Shoreham had a vent loss coefficient of 6.2, the peak drywell pressure calculated would be increased from 42 psig to approximately 49 psig.

However, a lower vent loss coefficient also leads to an increase in the energy deposition rate to the pool resulting in (1) higher water temperatures near the vent exit, (2) greater penetration of the steam jets into the pool water, and (3) potential dynamic effects. The potential dynamic effects include increased forces that could throw the pool water up against the deck and suppression chamber walls, possible vibration of the downcomer pipes, and possible water hammer effects. The magnitude and consequences of these dynamic effects are not known. The applicant has told us that he will consider dynamic effects during the detailed design effort and he has noted that he plans to add some form of structural constraint at the bottom end of the vent pipes to preclude the possibility of damage due to vibration or water hammer effects. The details of how this will be accomplished have not yet been determined. Items (1) and (2), discussed below, could lead to incomplete condensation of steam in the pool, and thus, higher drywell and wetwell pressures.

Complete steam condensation in the suppression pool depends on the pool water temperature, the energy addition rate into the pool, and the time it takes for the steam to reach the pool surface.

Shoreham will have a maximum bulk pool temperature of 149°F following the reactor blowdown. The highest bulk pool temperature observed in any of the Moss Landing tests was 163°F (test B-39).

Although Shoreham's bulk pool temperature will be well below that observed in test B-39, the local temperature near the vent pipe exists could be higher. Shoreham has predicted steam-plus-water flow rate of 456 pounds/sec per vent pipe while test B-39 had 325 pounds/sec per vent pipe. The diameter of the vent pipes is the same in both cases.

Comparisons with other Moss Landing tests show that some tests, such as B-16, have higher energy deposition rates (pounds of steam plus water/sec per vent pipe) than predicated for Shoreham, but at lower bulk pool temperatures. There are no test data at the combined bulk pool temperature and energy deposition rates that match Shoreham conditions.

In spite of the lack of these test data, we have concluded that steam condensation in the Shoreham pool will be complete. This conclusion is based on the following:

1. The energy addition rate per vent for Shoreham would be considerably less if the more realistic homogeneous blowdown model rather than the Moody model were used. Trial calculations using the homogeneous blowdown model gave results in essential agreement with the tests, whereas the results with the Moody model are always conservative.

2. As the suppression chamber becomes pressurized, the saturation temperature of the pool water will increase. This higher saturation temperature increases the likelihood of condensation. Test B-39 had a maximum pool saturation temperature of 225°F, while the 33 psig peak pressure of the Shoreham suppression chamber results in a saturation temperature of 256°F.

3. The greater depth of the Shoreham pool requires a long time for a steam bubble to reach the pool surface. The longer transit time would increase the probability of steam condensation.

4. Pressure suppression tests were conducted recently at Oak Ridge National Laboratory using a 1/10,000 scale model. Although the temperature range in the ORNL tests was between 80 to 130°F, their scaled mass flow rates were much higher than Shoreham's. Complete condensation was always observed in the ORNL tests. In addition, an empirical relationship between the steam mass velocity, the diameter of the vent pipe, and the distance the steam jet penetrates into the pool was developed. Based on the ORNL data, the steam that flows through the Shoreham vent system will condense within two feet of leaving the vent pipe.

5.2.3 Containment Design Pressure Requirements for BWR's

The Shoreham containment design pressure is 48 psig, while the conservatively calculated peak pressure is 42.2 psig. Thus, there is more than a 10 percent margin between the design and the peak pressure. As we have already discussed, the Shoreham containment design pressure has been conservatively calculated, and the containment critical parameters fall within the range of parameters investigated in the pressure suppression tests or are in the safe direction.

5.2.4 Conclusion

On the basis of the above considerations, and provided that the final structural design of the containment will preclude any significant bypass flow between the drywell and wetwell as discussed in the following section, we conclude that the functional design of the primary containment is satisfactory.

5.3 Structural Design of Primary Containment

From a structural viewpoint the major concern, developed by our review of the functional requirements, is the need to assure the structural and leaktight integrity of the floor separating the drywell from the pressure suppression chamber. The floor is to be designed for and tested to a 30 psig loading. It will not be monolithic with the walls of the primary containment or the reactor support sturcture. Flexible seals are to be provided between the drywell floor and the containment walls and the reactor support structure. We have reviewed the design criteria for the concrete floor. They require compliance with standard codes and practices and on this basis we have concluded that they are acceptable and should result in a design capable of resisting all applicable load combinations without serious cracking that might result in unacceptable bypassing of the suppression chamber.

The design of structurally acceptable seals, especially the one between the floor and the walls of the containment, is a more difficult problem. The applicant has submitted conceptual designs for each of these seals. In our opinion the concepts are feasible, but the practicality of developing acceptable designs remains in question. Since the information needed to resolve our concerns will be available only when the final designs are available we will require the applicant to submit the final designs of these seals for review and acceptance prior to construction.

The applicant has stated (Amendment 7, page V-2-23) that diagonal reinforcing rod will be used in the walls of the primary containment to resist seismic shear loads if test results do not indicate that aggregate interlock is adequate to resist these loads. The applicant, LILCO, is one of thre utility company sponsors of this test program, with Stone and Webster coordinating the program. The actual test work was done by Professor White at Cornell with consulting assistance by Hanson, Holly and Biggs. Test work has been completed and results were recently submitted to us informally. A report on the entire program will be submitted formally for our review in the near future. We expect to be able to give the Committee a preliminary report on our evaluation at the December meeting. The applicant is committed to use diagonal reinforcing rod if we do not conclude that the test results demonstrate that aggregate interlock is adequate to resist potential loadings. The construction schedule for the plant allows ample time for our evaluation before a decision on whether to use diagonal reinforcing rod must be made.

We have reviewed the design of the steel liner for the primary containment in the same manner and to the same depth generally employed for other designs. There are no unique structural problems raised by the configuration of the primary containment and we have concluded that the proposed designs of the primary containment liner and associated penetrations are acceptable.

We have reviewed the criteria proposed for the design of the reactor support structure. These criteria require consideration of all applicable loads and in our opinion should provide acceptable margins of safety. We have alerted the applicant of the importance of design details, such as the bolting specifications, but our evaluation of these areas must await completion of the final design.

The design leakage rate from the primary containment is stated to be 0.5% per day of the contained free volume. The structural design will, in our opinion, permit an appropriate leak rate test program to be developed at the operating license stage of review.

The proposed initial structural tests for the primary containment will demonstrate to the extent practical the adequacy of the design and the quality of the construction. These tests are consistent with those used to demonstrate the acceptablility of structures designed and constructed by methods and practices in general accord with those used for the Shoreham containment. For these reasons we consider these proposed initial tests to be acceptable. However, in view of the assurance required for the leaktight integrity of the floor and seals separating the drywell and suppression chamber, we will require periodic proof-testing of the complex. This may require that means be provided by design to enable the 2-foot-diameter vent pipes to be closed off during the periodic tests. We will require that an acceptable program be agreed upon in connection with resolution of the design of the floor mentioned above. The applicant has been advised of our position on this matter.

We have reviewed the proposed design criteria and preliminary designs for isolation valving for primary containment penetrations and find them acceptable with one exception. There are a large number (on the order of 100) of approximately 1 inch diameter instrument lines which penetrate the primary containment and which in some cases connect to the reactor coolant system. The applicant did not believe that the isolation valving on these lines would have to meet the criteria for isolation capability because of their relatively small size. He had intended therefore that the isolation capability on each of these lines would be provided by an excess flow check valve and a manual shutoff valve, both of which would be located outside of the primary containment. A failure of one of the lines that is connected to the reactor coolant system could cause an uncontained loss-of-coolant accident. A secondary failure or leak in any of these lines during an LOCA would breach the primary containment. Such a breach in containment could potentially result in there being insufficient NPSH for the ECCS pumps (see Section 6.1 of this report) so that, even for a relatively minor LOCA, the ECCS could not adequately cool the core. We have informed the applicant that the proposed isolation valving for instrumentation lines is not acceptable and that we intend to require isolation capability comparable to that provided on engineered safety feature penetrations. The applicant plans to submit a revised design, and we plant to make an oral report to the Committee on our evaluation of the new design.

5.4 <u>Post-LOCA Hydrogen Control</u>

The applicant has provided information on the evaluation and control of combustible gas in the post-LOCA period that is similar to that previously submitted in the applications for the Bell, Hatch, and Brunswick plants. It reflects an expectation of having results available from experimental work by the end of 1969 and completion of associated analytical studies on this problem by about mid 1970. The applicant has stated his intent to submit the results of this effort at that time.

Our review of the experimental program outlined by GE in connection with the radiolysis concern indicates the program to be generally acceptable. However, we understand that GE contractual arrangements with ORNL call for a six month period of dynamic loop tests using only distilled water. It is our opinion that the influence of coolant impurities that reasonably could be anticipated to exist prior to and during the post-accident period should also be explored. We intend to discuss this point with GE.

Our independent evaluation of the need for post-LOCA hydrogen control, which we intend to discuss in detail with the Committee in the near future, indicates to us that the radiolysis problem is more severe for the BWR than the PWR type of containment. In view of this, we have advised the applicant that, in our opinion, the accumulating information on this problem area is providing evidence of an increasingly conclusive nature that a valid safety concern exists. Further, consistent with the position taken on Diablo Canyon 2, we have informed the applicant that containment venting may not be acceptable as the primary means of post-LOCA hydrogen control and therefore that other systems may need to be developed to cope with the hydrogen accumulation problem to minimize, to the lowest practical extent, any exposure of the public. We will continue to try to resolve this problem area on subsequent plants and during the post-CP review for this plant.

5.5 Inerting of Primary Containment Atmosphere

The applicant contends that containment inerting is not necessary on the basis of information presented in GE Topical Reports APED 5454 and APED 5654. We have reviewed these reports and our conclusions and position thereon were presented to the Committee in our reports on Dresden 2 & 3. As discussed in these reports, we believe that inerting should be provided to increase the margin to allow for unanticipated increases in metal-water reaction and may also be of substantial benefit to the resolution of the radiolysis problem by extending the time to effect actions to cope with the gas evolution. We have therefore informed the applicant of our conclusion that the design must include provisions and equipment for inerting the containment. He will make provisions in the design and construction of the facility for the installation of this equipment (Amendment 6, page A.5-82).

We have concluded that this is acceptable at the construction permit stage for this plant.

5.6 Secondary Containment - Reactor Building

The reactor building provides controlled release of any radioactivity that leaks from the primary containment structure or that might be released during fuel handling. The building is cylindrical in shape, with reinforced concrete walls up to the polar crane rail. Above this, the structure is steel frame with insulated metal siding and metal deck roof. The entire building will be designed to withstand the effects of the design basis earthquake (0.2g) and tornado. It will be designed to have low in-leakage and out-leakage; all access opening, including the equipment door, will have air locks. All piping, ducting and electircal penetrations will be designed for low leakage with appropriate isolation valving. The building will be tested initially and periodically during the life of the plant to confirm that leakage is not greater than the design value, (50% of the building volume per day with a differential pressure of 0.5 inch of water).

We conclude that the design criteria proposed for the secondary containment building and its associated penetrations are acceptable.

5.7 Reactor Building Standby Ventilating System

Normal ventilation of the reactor building and ventilation in the event of an accident are provided by two completely independent systems. Automatic isolation of the reactor building and activation of the reactor building standby ventilating system, is provided in the event of an accident.

The reactor building standby ventilating system is automatically actuated by any one of the following signals:

1. High radioactivity level in the normal ventilation exhaust
2. High pressure in the primary containment
3. Low water level in the reactor
4. Rise in reactor building pressure toward atmospheric pressure
5. Manual initiation from the control room

The system is designated as an engineered safety feature and therefore is designed to seismic Class I and IEEE-279 criteria. Two fans, each of which provides 100% of the required capacity, are provided in the system.

The standby ventilation system in the Shoreham Plant provides not only absolute and charcoal filtering of the air discharged from the reactor building in the event of an accident, but also assures that any leakage from the primary containment will be mixed into the reactor building volume. For previous BWR plants, the calculated potential accident doses would be shown to be less than the 10 CFR 100 guidelines without consideration of building mixing because of the dilution provided by the elevated release. Since the

Shoreham facility does not have a stack, the assumptions made regarding building mixing have an important effect on the ability to limit potential doses for the fuel handling and DBA loss-of-coolant accidents to less than the 10 CFR 100 guidelines. The importance of mixing in the reactor building is not the result of the small additional dilution this may provide, but rather is the result of additional delay in release of radioactivity from the reactor building. The greatest effect appears in the 2-hour exclusion boundary dose.

The standby ventilation system in the Shoreham Plant is designed to preclude direct flow from any area in the reactor building directly into the discharge stream from the standby ventilation system. This is accomplished by drawing 30,000 cfm into a large number of intakes distributed throughout the 2×10^6 ft^3 volume of the reactor building. This 30,000 cfm flow is mixed by the fan and baffles in a mixing chamber. Of the total 30,000 cfm flow into the chamber, only 1/43 (700 cfm) is drawn into the standby ventilation system discharge stream. This stream passes through heating coils, prefilters, HEPA filters, and charcoal adsorbers before it is discharged from a vent on the roof of the reactor building. The rest of the 30,000 cfm ventilation flow is recirculated into the reactor building volume by a distributing duct system.

To conservatively estimate the effect that building mixing would have on accident dose calculations, the applicant has assumed that 1/43, the ratio of the discharge flow to the recirculating flow rate (700 cfm/30,000 cfm) of any radioactive release or leakage from the primary containment would pass directly into the discharge stream. The rest of the release or leakage is assumed to be carried into the recirculating flow, mixing with 50% of the building volume. Based on our analysis of the system we agree with the applicant that this is a conservative assessment of the effectiveness of the standby ventilation system.

In the accident analysis section of this report, we compare the calculated accident doses for the LOCA and fuel handling accident assuming no building mixing, the applicant's mixing model, and 100% building mixing.

It should be noted that in order for the LOCA fission products leaking from the primary containment into the reactor building to undergo building mixing and pass into the filtered discharge stream, it is necessary that bypass flow not occur through any of the lines which connect to the reactor coolant system or the primary containment and penetrate the reactor building. We have requested an analysis of this problem from the applicant and we will report on this to the Committee at the meeting.

6.0 ECCS AND OTHER ENGINEERED SAFETY FEATURES

6.1 Emergency Core Cooling System

The emergency core cooling system for the Shoreham Plant is essentially the same as that for the recently reviewed Hatch, Brunswick and Bell plants. The evaluation presented in Section 4.3 of our March 24, 1969 report to the Committee on the Bell plant is therefore also applicable to the Shoreham Plant.

The ECCS will be designed to performed properly if exposed to the fission product source term suggested in TID-14844. The applicant has provided his interpretation of this criterion (Amendment 8, Exhibit H, Item 1) which is consistent with that which we accepted on the Hatch and Brunswick applications.

An auto-relief interlock system will be included in the design to ensure that low pressure core cooling capability is available before auto-relief depressurization can be initiated by sensing pressure downstream from the core spray and LPCI pumps. This interlock system will meet the IEEE criteria. We consider this feature necessary and conclude that the proposed design criteria are acceptable.

In Amendment 7, Exhibit H. Item 6, the applicant states that in addition to leakage detection provisions, there will be remotely operable valves on the ECCS suction lines within ten feet of the penetrations into the primary containment wet well. Design and inspection of the piping between the containment and these valves will be in accordance with USAS B31.1 plus a limited number of additional requirements. A failure of this section of pipe during a LOCA could potentially result in the loss of all emergency core cooling capability.

In recent discussions with the applicant on this point he indicated that he is revising his design so that these isolation valves will be as close to the penetrations as possible (less than a foot). Also, he plans to raise the elevation of the ECCS pumps and their drives so that emergency core cooling capability would not be lost even if there were a failure in one of these short lengths of pipe or in one of the isolation valves, providing that such a failure did not occur until the pressure within the containment had been restored to essentially atmospheric pressure.

The applicant stated that he will submit the details of these revisions and their effects on accident analyses in the next amendment to his application. We will report to the Committee on our evaluation of this additio information at the meeting.

The Shoreham design, as in most recent BWR - vapor suppression plants, requires that there be some pressure in the primary containment in order that the ECCS pumps have sufficient NPSH, (ref. Amendment 6, Comment 6.2, pages A5-87). In a recent meeting with the applicant he indicated that the ECCS pumps to be installed in the Shoreham Plant will have significantly lower NPSH requirements than is presently indicated in the application. He has stated that he will submit the corrected values in the next amendment to his application. It appears, however, that even with these lower NPSH requirements there will still have to be some pressure in the primary containment, especially in the case of degraded ECCS performance, to assure adequate NPSH for the pumps. A failure to achieve or maintain leaktight integrity of the primary containment during a LOCA could therefore prevent adequate cooling of the core by the ECCS. Containment integrity, and therefore adequate NPSH could be lost, for example, by a failure of one of the non-isolable sections of the ECCS suction lines discussed above or the instrumentation lines discussed in Section 5.3.

We have discussed this problem with the applicant, and he has indicated that the NPSH requirements presently indicated in the application will be modified to reflect reduced requirements for the pumps which are actually to be procured. We therefore expect this issue to be resolved in essentially the same way it has been resolved in Duane Arnold and plan to report our conclusions to the Committee at the meeting.

The Shoreham ECCS includes a Standby Coolant Supply which consists of a permanently installed, normally valved off crosstie to the service water system. This would permit pumping water from Long Island Sound into the reactor vessel and primary containment in the unlikely event that the ECCS failed. Two keylocked valves which are operable from the control room, with a monitored intermediate drain, will be provided in the crosstie to protect against the unintentional introduction of salt water from the service water system into the ECCS. We have concluded that these provisions are adequate. During discussions with the applicant, he has indicated that he may propose that this Standby Coolant System could be used if there is inadequate NPSH for the ECCS pumps to perform properly.

Subject to the resolution of the ECCS suction lines and ECCS pump NPSH problems discussed above, we have concluded that the proposed design of the ECCS will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad-water reaction to less than one percent of the total clad mass, (3) terminate the temperature transient before the clad geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (4) reduce the core temperature and remove core heat for an extended period of time.

6.2 Other Engineered Safety Features

The Shoreham Plant will include main steam line flow restrictors and isolation valves, control rod velocity limiters and a control rod drive housing supports which are similar to those in other recent BWR plants. We reviewed these engineered safety features in detail on one or more of these other plants and concluded that they were conservatively designed to perform their intended functions. We have therefore also concluded that they are acceptable for the Shoreham Plant.

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

The instrumentation and control systems have been evaluated against the Commission's General Design Criteria (GDC) and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE-279) dated August 28, 1968. A comparative review was made with the Edwin I. Hatch Nuclear Plant, Unit 1 and the Brunswick Steam Electric Plant, Unit 1. The reactor protection instrumentation and control systems as well as the instrumentation which initiates and controls the engineered safety features were found to be functionally the same as those proposed and found acceptable in the above mentioned plants. The following discussion is limited to the BWR generic problem areas and to those areas for which new information has been obtained. Specifically, these areas are:

1. Auto-Relief System Interlock
2. Rod Block Monitor
3. Flow Reference Scram
4. Common Mode Failure Study
5. Single Failure Criterion

7.1 Auto-Relief System Interlock

The applicant will provide an interlock which will prevent the initiation of the auto-relief system unless the low pressure core cooling systems are available. The system to be provided will sense pressure downstream of the core spray and LPCI pumps and prevent auto-relief unless the pressure is above a preset value. The instrumentation and circuitry will be designed to meet the requirements of IEEE-279. This commitment and the design changes are identical to those proposed and accepted during our review of the Hatch and Brunswick applications.

7.2 Rod Block Monitor (RBM)

The applicant continues to believe that "the RBM system is installed only as an operational aid" and is, therefore, not required for safety. Our position is that the system is required for safety since we consider that spurious rod withdrawals are expected transients, and that expected transients should not result in fuel damage. As a result, the applicant has provided criteria which will be used to modify the design of the RBM to satisfy the requirements of IEEE-279. However, the applicant identifies the design features listed below as physical limitations which preclude complete compliance with these requirements:

a. A single pushbutton for rod selection will be used but redundant, isolated contacts will be provided.

b. The LPRM meter displays will be grouped in close proximity, but circuit isolation will be provided for each of the LPRM output signals.

c. A single rod selection acknowledge light for each rod will be provided.

d. Rod withdrawal block outputs from the RBM will be routed to a single cabinet for connection into the control rod drive control system.

e. A single switch will allow the bypassing of either RBM output to the manual control system.

These practical physical limitations are the result of the basic rod control system design and the spatial aspects of the protection required cf the RBM. Additionally, the grouping of indicators and/or combining of switching functions is made necessary in order to obtain a more effective and meaningful control board design from the human engineering standpoint.

It is our judgment that if the applicant's criteria (IEEE-279 with the above exceptions) governing the revision of the RBM system are properly implemented, the design will be acceptable. We will review the detailed design of the revised RBM system when completed in the second quarter of 1970.

The criteria and physical design limitations are identical to those identified in the Hatch and Brunswick applications.

7.3 Flow Referenced Scram

The applicant will provide a flow-referenced scram designed such that the power level flux trip point will be varied automatically as a function of recirculation flow. The circuitry and instrumentation for this change will be designed to satisfy the requirements of IEEE-279. This commitment and design change are identical to those obtained and accepted during our evaluation of the Hatch and Brunswick applications.

7.4 Common Mode Failure Study & Failure to Scram on Anticipated Transients Study

In Amendment 7, Exhibit H, the applicant provided a brief description of the study being conducted by the General Electric Company concerning the effect of common mode failures on the protection systems. The staff's Systematic Failures Status Report of August 4, 1969, provided comments on the future course of action and direction we plan to undertake with respect to this study.

GE is also conducting a study of the effects of a failure to scram in the event of anticipated transients, such as a turbine trip and of possible means of reducing the consequences. The results of this study will be considered in the final design of this plant.

7.5 Single Failure Criterion

Amendment 5 (Comment 7.2) of the application states that reactor protection systems and instrumentation systems which initiate or control engineered safety features, will be designed to comply with IEEE-279. However, other sections of the applicantion (e.g. page VII-7-13 of the PSAR) and discussions with the applicant indicate that some of these systems may not be designed to satisfy the single failure criterion of IEEE-279, but instead be designed to meet a "single component failure criterion". The seeming inconsistency is compounded by Amendment 7 which states that essential safety actions shall be carried out by equipment sufficiently redundant and independent that no single failure of an active component could prevent the required actions. Amendment y includes (page I-9-16a) defifinitions of "single failure", "active component" and "passive component", but these do not clarify the matter.

We have therefore requested the applicant to identify which reactor protection and instrumentation systems that initiate or control engineered safety features will, and which will not, be designed to meet the requirements of IEEE-279. The applicant has indicated that he will respond to this request in Amendment 9. We will report our evaluation of this information to the Committee at the meeting.

8.0 ELECTRIC POWER SYSTEMS

8.1 Offsite Power

The Shoreham Nuclear Power Station (SNPS) will be interconnected to the LILCO system through 138 kV and 69 kV circuits. Power from the unit's generator is fed via a single circuit containing the main step-up transformer and a circuit breaker to the 138 kV switchyard. The 138 kV switchyard is arranged in a two-bus configuration with circuit breakers and switches arranged to permit isolation and/or repair of either bus section. Four transmission circuits emanate from the switchyard (two per bus) each containing a circuit breaker at the connection to its respective bus. Two separate rights-of-way are provided, each containing two of the 138 kV circuits. The 69 kV circuit from the Wildwood substation enters the site sharing one of the aforementioned rights-of-way for a distance of one mile. This circuit, however, is mounted on separate towers and separated from the 138 kV circuits.

The equipment to provide offsite power to the Shoreham Plant satifies General Design Criterion 39.

The applicant has stated that stability studies indicate that the loss of this unit will not cause the interruption of offsite power to the engineered safety features.

Redundant, independent sources of offsite power are provided to power the engineered safety features upon loss of the normal unit supply. One source is derived from normal station service (NSS) transformer which is connected between the unit generator circuit breaker and the 138 kV switchyard. This design makes the NSS transformer independent of the main generator and allows it to be used during startup and shutdown of the unit. The second source is automatically made available from the reserve station service (RSS) transformer which is connnected to the 60 kV transmission circuit described above. Additionally, an onsite 55 MW gas-turbine generator will be available to supply auxiliary power to the RSS transformer in the event the 69 kV transmission circuit is out of service.

We conclude that sufficient redundant and independent sources of offsite power are provided to give reasonable assurance that no single failure will cause the loss of offsite power to the engineered safety features.

8.2 Onsite Power

The engineered safety feature loads are divided among three 4160 volt buses such that the operation of any two will supply minimum safety requirements. Three diesel generators are provided, each of which is exclusively assigned to one of three aforementioned 4160 volt buses. Each diesel generator has a continuous rating of 2500 kW. The diesel generators are started on loss of bus voltage or accident signals.

The emergency loads automatically connected to each of the three diesel generators are estimated to be 2,590, 2.620, and 2.590 kW. These loads exceed the continuous rating of the diesel generator, but will only occur for a short period of time (less than two hours). They do not exceed the 2,000 hour rating of 2850 kW. Although the generator loadings indicated are presently only estimated values, the applicant's criteria for diesel loading are to not exceed the continuous rating for long term requirements (beyond two hours) and not to exceed the 2,000 hour rating for short term requirements. We conclude that these criteria are acceptable.

The diesel generators and emergency buses will be located in separate rooms of a Class I building so that an incident in one diesel or bus will not involve another either physically or electrically. Each diesel generator will be provided with a day tank and a main fuel storage designed and located to meet Class I requirements. Each main fuel storage tank will contain sufficient fuel to operate its respective diesel fully loaded for seven days. The day tank will have a four-hour fuel capacity.

While redundant emergency buses are normally designed to be electrically independent of each other, 480 volt bus 106 will be provided with automatic transfer equipment to cause the tripping of the normal supply and effect the transfer to either 480 volt bus 104 or 105. Bus 106 supplies vital equipment such as the LPCI valves required for proper loop injection. The applicant has stated that the transfer equipment and instrumentation will be designed to meet the IEEE criteria. We consider that this commitment is satisfactory for the construction permit review.

Two dc systems will be provided. One system consists of two separate, redundant, and independent 125 volt batteries, each with its own charger and distribution board. Further, each battery will be located in a separate, ventilated room of a building designed to Class I seismic standards and the racks on which they are mounted will be designed to meet seismic requirements. The batteries will be sized to supply emergency loads for a minimum of two hours. Redundant emergency loads are divided between distribution boards and those which are not duplicated will be connected to buses with dual power supplies. The loss of any one battery will not preclude the operation of the minimum required engineered safety features. The second system consists of two separate, redundant, and independent 48 volt batteries and battery chargers. This system provi power to the source and intermediate range nuclear instruments and process radiation monitoring equipment. These batteries will each be located in a separate, ventilated room of a building designed to Class I seismic criteria and the racks on which they are mounted will also be designed to meet seismic requirements.

We have concluded that the proposed design of the onsite power system is acceptable.

8.3 Cable Design, Selection, Routing, and Identification

The applicant has documented his criteria for cable design, selection and routing. We conclude that if the criteria are followed, the probability of loss of redundant channels of protection from a single cause such as fire will be adequately low. The criteria for identification of safety related circuits are adequate.

8.4 Environmental Testing

The applicant has identified the electrical equipment, including cables, located within the primary containment which are required to operate during and subsequent to an accident. The qualifications test procedures and test conditions (simultaneous application of DBA conditions of temperature, pressure, and humidity) to be applied to obtain assurance that these components will perform as required have been adequately identified for the construction permit stage.

9.0 AUXILIARY SYSTEMS

9.1 Shutdown Cooling System

On shutdown, reactor steam will initially be blown down through the turbine bypass system to the main condenser, if available. If the main condenser is not available, the primary system relief valves will open automatically at their set pressure and vent reactor steam to the primary containment suppression pool. Feedwater will normally continue to be supplied to the reactor vessel by the regular feedpumps. If, for any reason the feedwater pumps are not available, such as would be the case if the main condenser is not available (since the feedwater pumps have condensing turbine drives), a low reactor water level signal will automatically initiate operation of the Reactor Core Isolation Cooling System (RCICS). Reactor steam is used in this system to pump water from the condensate storage tank into the reactor. Exhaust steam from the turbine is rejected to the suppression pool.

When the reactor coolant system pressure has been reduced to 35 psig (280°F saturated), operation of the reactor shutdown cooling system will be initiated manually. The reactor shutdown cooling system in the Shoreham Plant, as in most recent BWR plants, is a subsystem of the Reactor Core Residual Heat Removal System (RHRS). For shutdown cooling service, the two pumps and the heat exchanger in either one of the two RHRS loops will be used. Two isolation valves in series will be provided in the shutdown cooling lines to the RHRS to separate this low design pressure system (450 psig), which is outside of containment, from the high pressure in the reactor coolant system during normal operation. In those lines in which flow is into the reactor coolant system one of these valves is a check valve, but in the outlet line, both valves are necessarily externally operated valves. In response to our question (#4.4), the applicant has stated that in addition to being keylocked, these valves will have interlocks to prevent their being opened when the reactor coolant system pressure is greater than the design pressure of the RHRS. This interlock instrumentation will be designed to meet the single failure criterion.

On the basis of the design criteria and the preliminary design of the shutdown cooling system, we have concluded that the provisions for shutdown cooling of the reactor are satisfactory.

9.2 Auxiliary Cooling Water Systems

The auxiliary cooling water systems include:

1. Station service water system
2. RHRS service water system
3. Reactor Building Closed Loop Cooling Water System
4. Turbine Building Closed Loop Cooling Water System

The last of these, the Turbine Building Closed Loop Cooling Water System, does not provide cooling water to any critical components and systems. The first three do supply critical components and systems and therefore at least parts of these systems will be designed for the design basis earthquake.

The station service water system is a salt water system that provides cooling water for the heat exchangers for the emergency diesel generators, the Reactor Building Closed Loop Cooling Water System and a number of noncritical systems. The system has three half-capacity pumps, each of which can be powered by one of the diesel generators. We have confirmed that the design criteria and the preliminary design of the system are such that no single failure, including the rupture of any pipe, could incapacitate this system to the extent that it could not provide adequate cooling water for safe plant shutdown or to mitigate the consequences of accidents.

The Reactor Building Closed Loop Cooling Water System cools all the critical components and systems except the RHR heat exchangers. The system includes three pumps and two heat exchangers, which are cooled by the Station Service Water System. Two of these pumps and one heat exchanger are required for most normal modes of operation. For extended shutdowns and emergency conditions, only one pump and one heat exchanger are required. The system therefore has ample cooling capacity and redundancy of components. Furthermore, the system is subdivided into two isolable subloops, each of which can provide all essential cooling, so that no single failure, including a pipe rupture, could prevent the safe shutdown of the plant.

The RHRS Service Water System is an open salt water system that provides cooling only to the residual heat removal system (RHRS) heat exchangers for shutdown or mergency cooling. Normally, therefore, this system is not in operation. The system includes four half-capacity pumps, which are headered so that any combination of these pumps can supply either one of the two, 100% capacity each, heat exchangers. Adequate valving is provided to permit isolating any pipe rupture. Pressures in the RHRS Service Water System are lower than in the RHRS in any one of its several modes of operation, thereby precluding the possibility of salt water intrusion into the reactor or emergency core cooling systems. Radiation monitors are provided on the service water discharge lines from each RHRS heat exchanger.

Should a leak develop in a heat exchanger, it will be detected by one of these monitors and isolated.

On the basis of the considerations above, we have concluded that the proposed designs of the auxiliary cooling water systems are acceptable.

9.3 Radioactive Waste Systems

9.3.1 Liquid Radwaste System

The liquid radwaste system, collects, and segregates liquid wastes into three diferent types: radioactive low conductivity relatively clean waste, low radioactive but highly conductive contaminated wastes, and chemical wastes which are highly radioactive. After appropriate processing of each type, liquid wastes will be either discharged to Long Island Sound through the circulating water discharge pipe or shipped offsite for disposal.

The liquid radwaste system is the same as that for previous boiling water reactors with the exception that regenerant evaporators have been added to the system. The evaporators are a very desirable feature since they help convert much of the liquid radwaste into a form that can be shipped offsite for disposal, thereby reducing the amount of radioactivity released to the environs.

The system has regenerative demineralizers, but has the capability for processing and drumming the backwash from these demineralizers for offsite shipment. This design capability is a desirable feature even though the applicant currently intends to dispose of the backwash, when radioactivity levels will permit, by discharging it to Long Island Sound. Part 20 regulations will be met on an annual average basis at the condenser discharge.

Based on our review of the proposed design and the associated criteria, we conclude that the design of the liquid radwaste system is acceptable.

9.3.2 Gaseous Radwaste System

The gaseous radwaste system consists of an offgas holdup and decay system which will reduce waste gas radioactivity sufficiently to permit venting through a roof top vent. The offgas system will use a steam jet air ejector to exhaust the gases collected in the condenser into a catalytic recombiner. The catalytic recombiner is provided to recombine the radiolytic hydrogen and oxygen which constitues the vast bulk of non-condensable gases. This greatly reduces the volume of remaining gaseous radwaste. The remaining gases will normally be passed through five holdup tanks in series to provide a minimum of 10 hour holdup time.

The system is capable of individually pressurizing the holdup tanks thereby providing up to a 3 day holdup time. The minimum holdup time of 10 hours in the Shoreham Plant reduces total radioactivity release rates by a factor of about 6 below that for the 30 minutes delay time provided in most previous boiling water reactors. The 3 day holdup time for the Shoreham Plant produces releases decreased by a factor of about 28 below that for the 30 minutes delay associated with previous boiling water reactors.

The applicant has completed wind tunnel studies for normal operation with full ventilation flow which concludes that for the Shoreham site the plant, in effect, has an elevated release. We are not convinced that a wind tunnel study alone can be used to conclusively prove that the effluent releases from the plant vent will remain elevated under all possible conditions. In setting the technical specifications for this facility during the operating licensing phase of our review we will probably set a gaseous release limit based upon a ground level release, but we would be willing to change this limit if the applicant can demonstrate that based upon the actual operation of the offgas system, the releases do indeed remain elevated.

The difference between a stack release and a ground level release is approximately an order of magnitude in dilution. Hence, the increased offsite radioactive concentrations due to the lack of a stack and the decrease due to the increased holdup time have offsetting effects.

We have concluded that the design of the proposed gaseous radwaste system is such that gaseous waste discharges can be effectively controlled and that it is acceptable.

9.4 Fire Protection System

The fire protection system is not designated as seismic Class I equipment. As in other recent plants, we have taken the position that this is generally acceptable, but that any portions of the fire protection system whose failure could damage Class I structures or components must be designed to seismic Class I standards. We have notified the applicant of this position, and the applicant has stated orally that during final design of the system he will evaluate the possibility of damage due to failures of this system. If such possibilities exist, the fire protection system will be designed to seismic Class I standards.

9.5 Fuel Handling and Storage

The spent fuel storage pool is a steel lined, reinforced concrete tank. Test channels located behind every seam in the liner will lead to open telltale drains to detect any leakage of water. The pool cooling system includes two full-capacity pumps, two heat exchangers, and filter-demineralizers. Connections to the RHR system provide for additional cooling capacity and make-up water.

Both administrative procedures and design features will be provided to limit the consequences and probability of releasing radioactivity from the spent fuel. During refueling the reactor building airlock doors will be closed.

Analyses of damage to the pool in the event that the spent fuel cask is dropped have not been initiated by the applicant. We have informed the applicant that such analyses must be done during detailed design with the objective of showing that the resulting damage would not cause gross loss of spent fuel cooling capability or loss of operability of vital equipment that could conceivably be flooded by a major pool leak.

10.0 STRUCTURAL DESIGN (OTHER THAN THE PRIMARY CONTAINMENT)

The applicant has stated that structures and equipment whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the station and the removal of decay and sensible heat are defined as Class I for purposes of seismic design. Structures and equipment which may be essential to the operation of the station, but which are not essential to a safe shutdown are considered Class II. We have reviewed the applicant's detailed listing to determine whether all structures, systems and components are being considered in the appropriate classification. We have concluded that all structures, systems and components have been classified correctly except the steam-power conversion system (i.e. the main steam line, turbine condensate and feedwater systems). With the applicant's definition, these components would have to be considered Class I. However, considering the intent of this classification and other factors involved, we have concluded that these steam-power conversion system components need not be classified Class I. The main steam line is being treated as a special case as discussed in Section 4.4. All other structures systems and components have been correctly classified. We have also reviewed the design criteria for Class I structures and components to confirm that appropriate loadings and stress and/or strain limitations will be considered.

As discussed in Section 2.4, Class I items will be designed for a DBE of 0.20g, and an OBE of 0.10g maximum horizontal ground accelerations. We and our seismic design consultants have reviewed the general design criteria and methods that the applicant proposes to use for the design of Class I structures, systems, and components. We have concluded that these criteria and methods are acceptable.

The applicant intends to design Class II structures to stresses 1/3 above working stresses when operating loads are combined with OBE or wind loads. Although the turbine building is listed as Class II structure, the applicant has informed us that the gaseous radwaste treatment area which is located in the turbine building, will be designed to Class I standards. These approaches are acceptable.

The Shorehcm facility will be designed for tornado loadings that are in accord with our current requirements, (300 mph wind speed with a 60 mph translational speed, 3 psi pressure drop in 3 sec). We have reviewed the proposed design approach and have concluded that it is acceptable. A strong motion seismograph will be located at foundation level inside the reactor building.

The applicant had originally proposed to use a Cadweld splice testing program which is not acceptable to us. In recent meetings, he has indicated that he will probably modify his program to satisfy our requirements. We therefore expect to be able to report at the ACRS meeting that this matter has been resolved.

On the basis that the program for Cadweld splice testing is resolved to our satisfaction, we have concluded that the structural design criteria and preliminary structural designs are acceptable.

11.0 ACCIDENT ANALYSIS

We have evaluated the potential radiological consequences for the same design basis accidents as those considered for other recent BWR plants -- control rod drop, fuel handling, steam line break, reactor coolant system pipe break (LOCA) and a gas decay tank rupture. The assumptions used in calculating the potential doses for each accident are essentially the same as those used on previous plants and are listed in Appendix B. The results of our analyses, and the comparable values calculated by the applicant are presented in Table 11.1. The doses reported by the applicant in the PSAR are indicated on the table in parentheses. Only the values he calculated "using AEC-DRL assumptions" are indicated. The applicant also reported dose values for each accident which he calculated using his "design basis assumptions". In every case, these doses are significantly lower (two or more orders of magnitude) The differences in assumptions are primarily the differences in source terms and decontamination factors identified previously on other BWR plants.

As discussed in Section 5.6, the Shoreham design includes special provisions to assure that any radioactive release within the reactor building would be mixed into the building volume before being released to the environs by the standby ventilation system. In order to demonstrate the effect on offsite doses of such mixing, we have considered three cases. Case A assumes no mixing, which means that any radioactivity release passes directly into the discharge stream from the ventilation system. Case B assumes 100% mixing throughout the building volume and Case C assumes only 50% of the building volume is utilized for mixing. As discussed in Section 5.6, we have concluded that the 50% mixing model is a conservative assumption for purposes of calculating potential offsite doses.

The doses indicated in Table 11.1 for the reactor coolant system pipe break accident (LOCA) were calculated assuming no reactor building bypass flow, i.e., all radioactive leakage from the primary containment is assumed to be processed by the standby ventilation system. As discussed in Section 5.6, we are still working with the applicant to confirm the validity of this assumption and will report our final conclusions on this to the Committee at the meeting. Provided that a sound basis for this assumption can be established, it can be seen from Table 11.1 that the calculated potential offsite doses for all accidents are below the guideline values in 10 CFR 100.

OFFICIAL USE ONLY

TABLE 11.1

POTENTIAL DOSES DUE TO DESIGN BASIS ACCIDENTS

	2 Hour Doses in Rem at 1000 Feet Exclusion Distance		Course of Accident Doses in Rem Five Mile LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
1. Control Rod Drop	31 (34)*	0.8 (2.1)	5	0.1
2. Fuel Handling Accident				
Case A	546	10	14	0.3
Case B	46	0.8	11	0.12
Case C	58 (29)	1.1 (5)	11 (3.3)	0.12 (0.1)
3. Steam Line Break	55 (20)	0.2 (0.01)	1.4	0.1
4. Reactor Coolant System				
Pipe Break (LOCA)**				
Case A	2359	93	373	8
Case B	49	19	155	2.0
Case C	153 (60)	6 (8.1)	169 (46)	2.4 (2.4)
5. Gas Decay Tank Rupture	--	10	--	0.2

Case A - No building mixing
Case B - 100% building mixing
Case C - Applicant-Staff mixing model (50%)

* Doses in parenthesis are those calculated by applicant "using AEC-DRL assumptions."
** Assumes no reactor building bypass flow.

OFFICIAL USE ONLY

12.0 QUALITY ASSURANCE

We have reviewed the quality assurance program plan presented by the applicant for the design, construction and operation of the Shoreham Nuclear Power Station with regard to the applicant's stated objective of meeting the intent of the AEC proposed "Nuclear Power Plant Quality Assurance Criteria," Appendix B of 10 CFR 50. The QA program plan is presented in PSAR Volume I, Section 1 (Rev 4/4/69), and Volume III, Appendix E (Rev 8/15/69).

The applicant proposes to establish a Quality Assurance Organization within the company specifically for the Shoreham Nuclear Power Station. An experienced graduate engineer with a broad power engineering background has been appointed as Quality Assurance Administrator (QAA) for this project. He will report to the LILCO Shoreham Project Manager and will be responsible for the development and execution of the overall Quality Assurance Program. The LILCO QAA will prepare a Quality Assurance Manual covering the entire QA program. LILCO will delegate appropriate phases of the QA program to Stone and Webster (S&W), the engineer-construction manager, and GE, the NSSS supplier; therefore certain of these contractors' QA and QC procedures will constitute an integral part of the overall QA program, even though they may appear in the QA manual only by reference. Although S&W is precluded from doing any of the actual construction of the facility, as construction manager S&W will perform much of the QA/QC effort during construction.

The LILCO QAA will audit the Stone and Webster and General Electric phases of the QA program and maintain liason with the S&W Coordinator and the GE Project Manager. He will be assisted by his own staff in the office and at the site, as well as by engineers from the various disciplines in the LILCO organization as required.

The LILCO QAA will report to the Shoreham Project Manager rather than to a LILCO Vice President. We believe this arrangement will prove satisfactory in this case because of the qualifications of the persons involved; however, through the Division of Compliance inspection we plan to give particular attention to the relationship between the QAA and the Project Manager to assure proper independence is maintained.

The LILCO Shoreham Quality Assurance Program plan is very similar to the plans for other nuclear power plant projects in which S&W has been involved and which we have found to be in reasonable conformance with the proposed QA criteria.

The S&W QA Coordinator for Shoreham reports organizationally to the S&W Project Manager for Shoreham; however he has a line of communication with the QA Manager for the entire S&W organization. In addition, S&W has an Engineering Assurance Review Committee which is concerned with QA aspects of the Shoreham project. It is comprised of engineers from the various disciplines in S&W and reports to the S&W Chief Engineer and to the QA Coordinator for Shoreham.

Stone and Webster will maintain both a field quality control organization at the site and a vendor shop quality control staff. The latter will be under the jurisdication of the S&W Chief Quality Control Inspector who communicates with LILCO via the S&W Coordinator for Shoreham.

The General Electric Company has an integrated Quality Assurance organization which has been established to handle all of their BWR projects. It maintains contact on quality assurance matters on a particular project through the GE Project Manager in their Nuclear Systems Projects and Procurement Section of APED. We have had the opportunity to investigate the GE QA organization in connection with other recent BWR projects and believe that organizationally and functionally it conforms reasonably well with the intent of the proposed AEC QA criteria.

We conclude that with proper development and implementation of the LILCO Quality Assurance Program plan for the Shoreham Nuclear Power Station as presented in the PSAR, the intent of the proposed "Nuclear Power Plant Quality Assurance Criteria" Appendix B of the 10 CFR 50 will be met in the design, construction and operation of this station.

13.0 CONFORMANCE WITH GENERAL DESIGN CRITERIA

We have evaluated the design criteria and the preliminary design of the proposed Shoreham Plant with reference to the 70 General Design Criteria and found no indications that the intent of any of the General Design Criteria will not be met by the final design of the plant.

14.0 Technical Qualifications and Conduct of Operations

14.1 Technical Qualifications

Long Island Lighting Company (LILCO) will be responsible for the overall design, construction, and operation of the Shoreham Nuclear Power Station Unit No. 1. Stone and Webster Engineering Corporation has been engaged to act as Long Island Lighting Company's agent, with direct responsibility for design, and construction management. General Electric Company is the nuclear steam supply system supplier.

The applicant has had extensive experience in the design, construction and operation of fossil-fueled electric power stations. Although this is the company's first nuclear station, LILCO has participated in the research and development actities of Atomic Power Development Associates (APDA), Power Reactor Development Company (PRDC), and Empire State Atomic Development Associates (ESADA).

Long Island Lighting Company's engineering staff is composed of approximately 90 graduate engineers divided into four divisions, one of which is the nuclear engineering division. At least five of the engineers in this latter division have earned Master of Science degrees in Nuclear Engineering. Other disciplines traditionally associated with the utility industry are appropriately represented on the engineering staff.

Stone and Webster Engineering Corporation has been actively engaged in nuclear engineering and the construction of nuclear plants since 1954. They have participated in the design and construction of six completed nuclear stations and have seven more under design or construction.

General Electric Company has been actively engaged in the development, design, construction and operation of boiling water reactors since 1955 including at least ten operating BWR's and have fourteen others under construction in the U.S.

On the basis of the above considerations and our contact with project personnel during our review, we have concluded that the applicant and his contractors collectively are technically qualified to design and construct the proposed Shoreham Station Power Plant Unit No. 1.

14.2 Operating Organization and Training

The applicant's proposed station organization consists of a Station Manager assisted by a Chief Engineer and three organizational groups; the maintenance group under the Maintenance Engineer; a Reactor Engineer, Instrument and Controls Engineer, and the Radiation Protection and Chemical Engineer and their associated technicians all reporting to the Results Engineer; and five 4-man operating shifts reporting to the Operations Engineer.

The applicant plans to train his operating staff in the same program used at other recent General Electric BWR's. The training program is broken up into five basic parts:

(1) Basic nuclear course,

(2) BWR Technology course,

(3) BWR Operator Training,

(4) Specialist Training, and

(5) On-site training.

We conclude that the applicant's proposed plans for his operating organizational structure and training program are generally satisfactory except that the proposed operating crew size of four men is one man less than we consider acceptable at this time. We have informed the applicant of our concern and urged him to reconsider his proposal. We shall review this subject in detail at the operating license stage.

14.3 Conduct of Operations

The applicant has identified the major items that he will include in his plans for emergency preparedness, operating procedures, review and audit of station operations and his preoperational and initial startup program.

We consider that these plans are adequate for the construction permit stage. They will be reviewed in detail at the operating license stage.

15.0 RESEARCH AND DEVELOPMENT PROGRAMS

A number of areas requiring further analytical, experimental, design development, or testing efforts to substantiate the adequacy of a system design or safety feature of second-generation boiling-water reactors similar to the Hatch Plant have been identified during the course of previous reviews. These are discussed in Section 9.0 of our March 29, 1969 report to the Committee on the Bell Station, and that discussion applies equally to Shoreham.

The following programs are included:

1. Core spray and core flooding heat transfer effectiveness in a full-scale boiling water reactor
2. Steam line isolation valve closure time testing under accident conditions
3. Effects of fuel rod failure on ECCS performance
4. Effects of fuel bundle flow blockage on cooling capability
5. Verification of fuel damage limit criterion
6. Effects of cladding temperature and clad material on ECCS performance
7. Verification that the analytical model used to predict the ability of HPCI to depressurize the reactor is conservative.

Information on the program being sponsored by GE obtained subsequent to our report on the Bell Station and our independent evaluation of the need for post-LOCA hydrogen control is discussed in Section 5.4 of this report. We are satisfied that the specified program concerning radiolytic decomposition, in addition to the industry-wide efforts presently underway to obtain satisfactory resolution of this issue, will be concluded prior to operation of the Shoreham Plant, more probably by early 1970.

Based on our review of the research and development programs proposed, we conclude that these programs are timely and are reasonably designed to accomplish their respective development objectives, will provide adequate information on which to base analyses of the design and performance, and should lead to acceptable designs for the respective systems.

16.0 CONCLUSION

As we discussed in this report in the Sections indicated below, the following have been identified as items that will be resolved during the construction of the plant. Sufficient preliminary information is available on each of these items to indicate that they can be resolved satisfactorily, or that acceptable alternate solutions exist.

1. Peak storm surge for which the plant will be designed - Section 2.3 on Hydrology
2. Inservice inspection program - Section 4.6 on Inservice Inspection
3. Detail design of primary containment floor seals - Section 5.3 on Structural Design of Primary Containment
4. Use of diagonal reinforcing rods in walls of primary containment - Section 5.3 on Structural Design of Primary Containment
5. Provision for the control of post-LOCA hydrogen - Section 5.4 on Post-LOCA Hydrogen Control
6. Capability of design of accommodate common mode failures - Section 7.4
7. Capability of design to accommodate a failure to scram in the event of an anticipated transient - Section 7.4

A number of items have also been identified which are presently unresolved, but on which we expect to be able to report resolution at the Committee's December meeting. These items, and the Sections in this report in which they are discussed are as follows.

1. Inspection of main steam piping - Section 4.4 on Main Steam Piping
2. Isolation valving for instrumentation lines that penetrate primary containment - Section 5.3 on Structural Design of Primary Containment
3. Potential for bypassing Standby Ventilation System - Section 5.6 on Reactor Building Standby Ventilation System
4. Design provisions to accommodate failure of ECCS suction lines - Section 6.1 on Emergency Core Cooling System
5. NPSH requirements for ECCS pumps - Section 6.1 on Emergency Core Cooling System
6. Applicability of IEEE criteria to instrumentation and control systems for engineered safety features - Section 7.5 on Single Failure Criterion
7. Amount of testing of Cadweld splices - Section 10.0 on Structural Design

Subject to satisfactory resolution of the items above, we conclude that the proposed Shoreham facility can be operated at the proposed site without undue risk to the health and safety of the public.

APPENDIX A

PROBABILITY OF AN AIRCRAFT CRASH AT THE SHOREHAM SITE

The Long Island Lighting Company (LILCO) Shoreham Nuclear Power Station (SNPS) is to be located 4.75 miles from the Grumman Aircraft Company (Peconic River) Airport and about 1/2 mile off a straight line projection of one of the airport's two runways. This Appendix discusses the relative probability of an aircraft crash as a function of distance from the airport and describes the calculational methods used and results obtained by both the applicant and by the staff.

The types of aircraft using the Grumman Aircraft during 1968 can be categorized as follows:

TYPES	PERCENT OF TRAFFIC
1. Transport (Air Carrier)	63
2. Grumman	20
3. Military	14
4. Miscellaneous	3

DISCUSSION OF APPLICANT'S ANALYSIS

The applicant based his analysis entirely on air carrier (transport) statistics. We have compared the data on aircraft crashes near airports submitted in Amendment No. 3 with information from the National Transportation Safety Board (NTSB), the Federal Aviation Agency (FAA), and the Metropolitan Edison Company's Three Mile Island Unit 1 application, and have concluded that the data are substantially correct. These data include all airport-related fatal crashes of transport air carriers occurring within ten miles of an airport for the period 1956-1965. Transport air carriers include all commercial and passenger aircraft. Military aircraft, small private aircraft, training flights, helicopters and planes under test are not included. Data relating to transport air carriers were used because they seemed to represent the largest percentage of any type aircraft using the Grumman Airport and because this is the group for which the best crash data are available.

The applicant's analysis considers only data for fatal crashes. The reasons given for this limitation are that fatal crashes are generally high-energy impact crashes, whereas nonfatal crashes are often semicontrolled crashes so that a known sensitive area (and large structure) such as a reactor facility could probably be avoided.

Using these data, the applicant obtains a curve (Figure 14 of Amendment 3) of the crash probability with distance from the end of a runway. The curve indicates a rapidly decreasing crash probability with distance from the runway so that at distances beyond about five miles the curve is shown as having leveled off to a constant, "low frequency region" value. On the basis of this curve, the applicant postulates that the probability he evaluates for a crash at the Shoreham site (0.65×10^{-6} per year) is essentially equal to the average value of the low frequency region (0.3×10^{-6} per year). Furthermore, the probability of a crash at the site is essentially equal to the average probability of a crash determined for the entire region within ten miles of an airport, (0.47×10^{-6} per year). The applicant therefore maintains that there appears to be essentially no runway path orientation for crashes at the distance of the Shoreham site from the runway (4.75 miles). On this basis, the applicant concludes that no unusual provisions in the design or operation of the Shoreham Plant need to be made to accommodate aircraft crashes.

In evaluating the applicant's analysis, it is important to realize that the data used include only one crash for distances greater than four miles from the end of the runway because of the geometry used in the probability calculations (see Figure A-2). The applicant used this single crash to evaluate an average crash probability for the "low frequency region", (five to ten miles). The crash probability curve was then plotted by drawing a smooth curve through calculated probabilities for distances less than four miles using the computed probability for the low frequency region as an asymptote.

We investigated the significance of the assumption that only fatal crash data were applicable by considering both total (fatal and nonfatal) crash data and fatal crash data alone for the year 1965. Figure A-1 shows how the crash density (number of crashes per square mile) varies with the distance from the airport for these two groups of data. From this figure it can be seen that the crash density decreases more rapidly when considering total crashes than when considering only fatal crashes; hence, the use of only fatal crashes is slightly more conservative in determining the effect that the proximity of an airport has on the relative probability of an aircraft crashing at any given location. An analysis of the data for the year 1966 gave similar results.

In our April 4, 1968, Report No. 3 to the ACRS on Three Mile Island Unit 1, we indicated that the probability of a crash on a plant in the background region, i.e. in areas well away from any airport, is approximately 1×10^{-7} per year, i.e one crash every 10,000,000 years. In this calculation, a value of 50 overflights per day for a 20-mile wide flight corridor was assumed. This "background" crash probability is directly proportional to the number of overflights per day and inversely proportional to the width of the flight corridor assumed. The Washington-Boston-New York air corridor, which is about 50 miles wide, has approximately 500 overflights per day and hence the computed crash probability in this area is increased to about 4×10^{-7} per year. In our Metropolitan Edison report we also suggested that we should probably assign an uncertainty factor of ten in the crash density in order to allow for uncertainties in many of the assumptions. Hence, the crash probability in the Washington-Boston-New York air corridor could lie anywhere between 4×10^{-6} and 4×10^{-8} per year.

Because of the large uncertainties involved and the limited data available we have concluded that the absolute values of crash probabilities cannot be stated with a high degree of confidence. We have also, concluded, however, that the relative values of probabilities evaluated using consistent assumptions are meaningful and provide a basis for evaluation of reactor sites near airports. Using the same data as the applicant (transport statistics), we therefore calculated the probability of aircraft crashes as a function of distance from an airport by several different methods, but normalized the results so as to eliminate the differences in absolute values of the probabilities obtained by the different methods and retain the relative shapes of the curves.

We calculated the variation of crash probability with distance by using several different geometries. We considered concentric circles around the airport, neglecting runway orientation; we considered a 60° fan-shaped path symmetric about the extended centerline of the runway, and we considered a path one mile wide for the first two miles opening into a 90° angle at the two-mile distance (see Figure A-2). The normalized plots of the crash probabilities for these three geometries as well as for the applicant's geometry are shown in Figure A-3. For all geometries we have essentially the same exponentially-shaped curves. The general appearance of such an exponential curve plotted on linear coordinates, and particularly the point at which the curve appears to have essentially attained its asymptotic value are, of course, very dependent upon the scale used for the coordinates of the graph.

Figure A-4 is a plot of the same data on semilog coordinates. The straight line plots show that each curve is composed of two exponential components. In each case, there is a significant change in the slope of the semilog plot at about two miles from the end of the runway. The continuous straight-line plot at distances greater than two miles indicates that the crash probability continues to approach a "background" value. As noted above, the background value is not a single number but an uncertain probability band and therefore the exact distance which is out of the influence of the airport is not certain.

The same effect can be seen on Figure A-5 which is a histogram of the percent of fatal crashes which occurred at various distances (out to ten miles) from airports over a ten-year period. These data were analyzed for the specific geometry of the runways at the Peconic River Airport near the proposed Shoreham Plant, but the general shape of the crash-distribution histogram would not change significantly for other runway configurations. One crash is equivalent to 2.2% on the graph.

As shown in Figure A-5, 85% of the crashes which occurred within ten miles of an airport occurred within two miles of the airport. Crashes which occurred on the airport were excluded. Seven percent occurred between two and four miles and each additional mile thereafter adds about two percent.

As a result of our examination of the histogram (Figure A-5) and the crash probability data (Figures A-3 and A-4), we conclude that there is little change in the crash probability after about two miles regardless of the method of analysis used. We note, however, that there is a sharp increase in crash frequency within the two-mile distance.

SPECIFIC TYPES OF AIRCRAFT USING THE GRUMMAN AIRPORT

We also investigated the possible effects on the probability of crashes for each of the specific kinds of aircraft and aircraft activity characteristic of the Grumman Airport.

Transport Type Flights

The "transport" or "air carrier" type activity at the Grumman Airport is divided approximately equally between commercial transport flights and airline training flights. The training flights include landing and takeoff training as well as contingency conditions such as engine and other component failures.

In order to determine what effect this special type of use may have on the probability of a crash, we have examined reports[1] which compare the accident rates for all types of training flights to those for normal transport flights for the four year period from 1964 to 1967. These reports indicate that the number of accidents involving fatalities per hour of aircraft

---

1 Bureau of Aviation Safety Reports, General Aviation Accidents, A Statistical Review, Civil Aeronautics Board and National Transportation Safety Board, Washington, D. C.

flying time for instructional and training flights is only about 1/2 that associated with commercial flying. Recognizing that an "hour of flying time" for training flights usually represents more landings and takeoffs (movements) than an "hour of flying time" for commercial flights. We have concluded that our previous analysis of crash probabilities in the vicinity of an airport using crash statistics for air carrier flights to represent all flights in the transport category is a conservative method of analysis.

The applicant stated in Amendment 8 that recent data on aircraft movements at the Grumman Airport in the months of June, July, and August, 1969, indicates a substantial reduction in the use of the airport for training flights. The applicant has suggested that this reduction in training flight activity is due to two reasons: (1) these training flights generally originate at the New York City commercial airports which are very congested and often have long ground delays. They are therefore being relocated to other areas in the country having less air traffic, and (2) actual training flights are being replaced by the use of flight simulators.

Grumman Flights

The aircraft activity classified as "Grumman" is comprised of research and development aircraft and first flight testing of Grumman production aircraft. We have obtained data from the National Transportation Safety instructional and training flights is only about 1/2 that associated with commercial flying. Recognizing that an "hour of flying time" for training flights usually represents more landings and takeoffs (movements) than an "hour of flying time" for commercial flights, these statistics indicate that a training flight movement is safer than a commercial flight movement. We have therefore concluded that our previous analysis of crash probabilities in the vicinity of an airport using crash statistics for air carrier flights to represent all flights in the transport category is a conservative method of analysis.

The applicant stated in Amendment 8 that recent data on aircraft movements at the Grumman Airport in the months of June, July, and August, 1969, indicates a substantial reduction in the use of the airport for training flights. The applicant has suggested that this reduction in training flight activity is due to two reasons: (1) these training flights generally originate at the New York City commercial airports which are very congested and often have long ground delays. They are therefore being relocated to other areas in the country having less air traffic, and (2) actual training flights are being replaced by the use of flight simulators.

Grumman Flights

The aircraft activity classified as "Grumman" is comprised of research and development aircraft and first flight testing of Grumman production aircraft. We have obtained data from the National Transportation Safety

Board for aircraft crashes involving experimental aircraft for the period 1964-1968. Although we cannot calculate absolute crash probabilities using these data because data on the total number of experimental aircraft movements are not available, we can determine how the crash probability for experimental aircraft varies with distance from an airport. Figure A-6 is a histogram demonstrating the variation of crash density (number of crashes per sq. mi.) with distance from the end of a runway for this five year period. From this curve, we conclude that the crash probability for experimental aircraft at distances from an airport comparable to the distance for the Shoreham site from the Grumman Airport is essentially that associated with general overflights of experimental aircraft. We therefore believe that the Shoreham site is sufficiently distant from the Grumman Airport that the crash probability associated with experimental aircraft has essentially decreased to its background value for the Long Island area.

In Amendment 8, the applicant has observed that experimental flights are made only after the aircraft have undergone thorough inspections, that these flights are made by experienced engineer-pilots, and that the actual testing is done either over the ocean or over other areas remote to the airport rather than near the airport itself. In addition, many of the experimental flights performed at the Grumman airport are for testing auxiliary equipment on board the aircraft rather than for testing the aircraft engine or airframe itself.

Military Flights

The aircraft of this type activity are primarily Grumman aircraft which have been produced for the military (chiefly for the Navy). We have obtained reports from the USAF[2] and the USN/USMC[3] which summarize military aircraft accidents in the vicinity of airfields over five year periods. Although we were again unable to calculate absolute crash probabilities from these statistics because the total number of military aircraft movements per year is not available, we have analyzed these data and have determined how the crash probabilities for military aircraft vary as a function of distance from an airport. Again we conclude that at distances comparable to the distance of the Shoreham facility from the Grumman Airport, the crash probability associated with military aircraft approximates the background crash probability for this type activity.

Miscellaneous Flights

The 3% miscellaneous flight activity at the Grumman Airport is composed primarily of small aircraft (usually private) and helicopters. A crash of this type aircraft would therefore be a low-energy impact. We have been unable to obtain any data specifying the location of crashes relative to an airport for this type of activity. Owing to the small percentage of aircraft in this category and the small size of the aircraft, we do not believe that

---

[2] USAF Aircraft Accidents in Vicinity of Airfields, 5 mile Zone 1960-1964 (Study NR 21-65) Directorate of Aerospace Safety, Deputy the Inspector General, USAF, Norton Air Force Base, California

[3] Summary of Aircraft Accidents within 5 miles of USN/USMC Airfields FY 1964-1968, Project Study Group 68-13, Aircraft Analysis Division, Naval Safety Center, Naval Air Station, Norfolk, Virginia.

any peculiarity in the crash characteristics of this type of aircraft could effect our conclusion regarding aircraft crash protection at the proposed Shoreham site.

CONCLUSION

We have examined the probability of an aircraft crash at the Shoreham site by separately analyzing crash statistics for each of the various types of activity at the Grumman Airport and have determined the effect of the calculational technique used on the crash probability. Based upon these analyses we have concluded that the proposed site is sufficiently far away from the Grumman Airport that the proposed plant need not be designed or operated with special provisions to protect the facility against the effects of an aircraft crash.

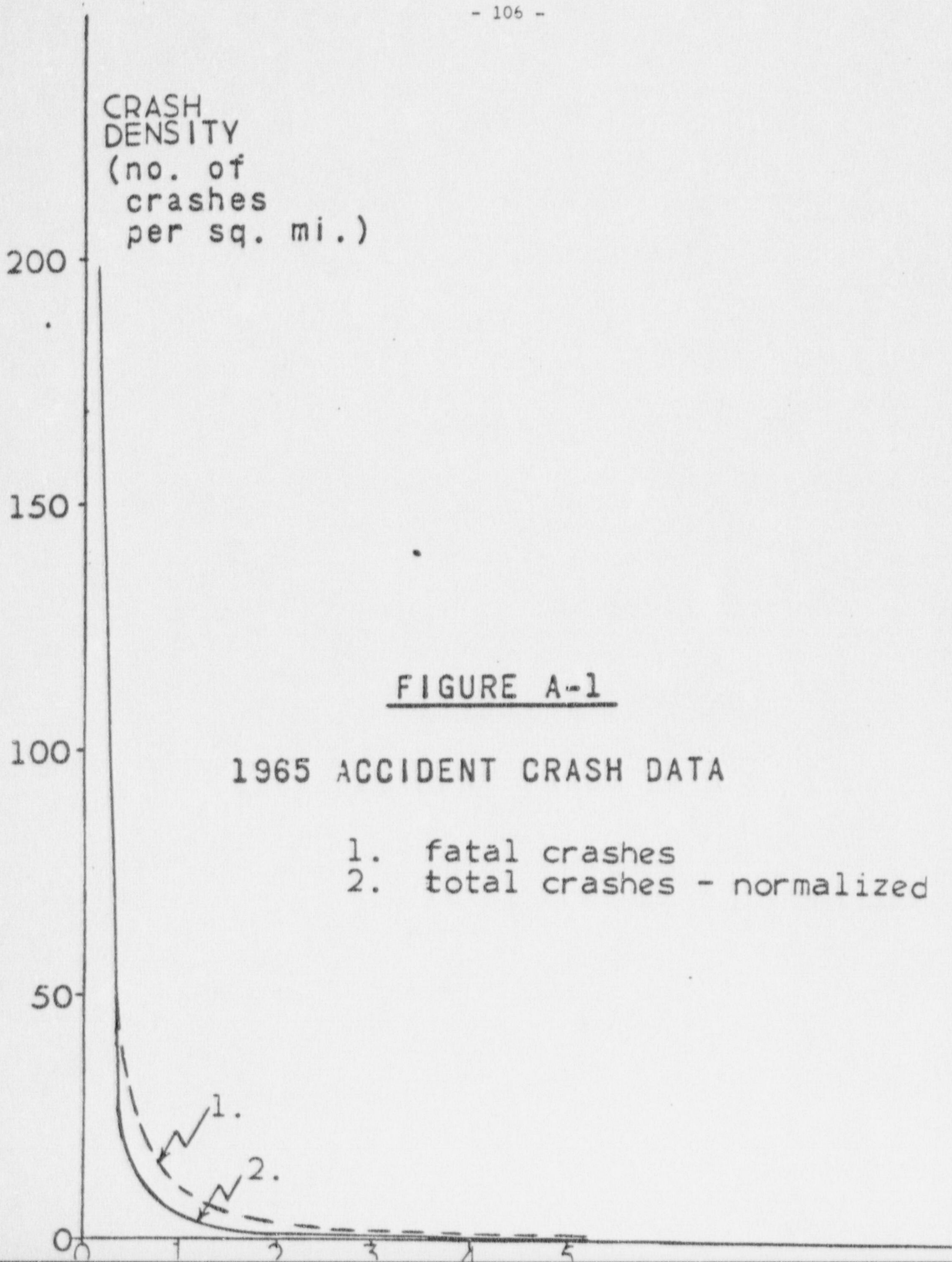

FIGURE A-1

1965 ACCIDENT CRASH DATA

1. fatal crashes
2. total crashes - normalized

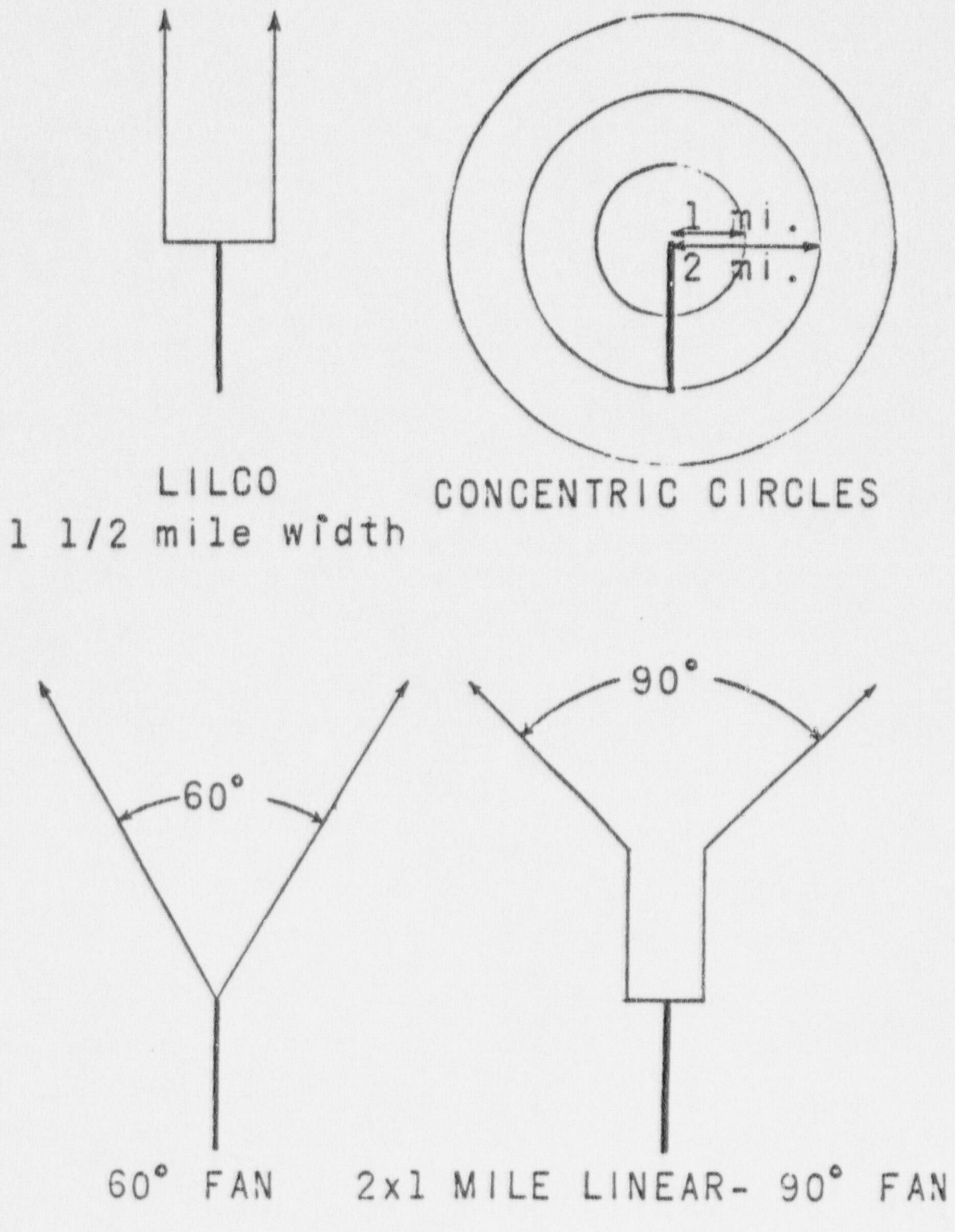

FIGURE A-2

VARIOUS FLIGHT PATHS USED IN CALCULATIONS

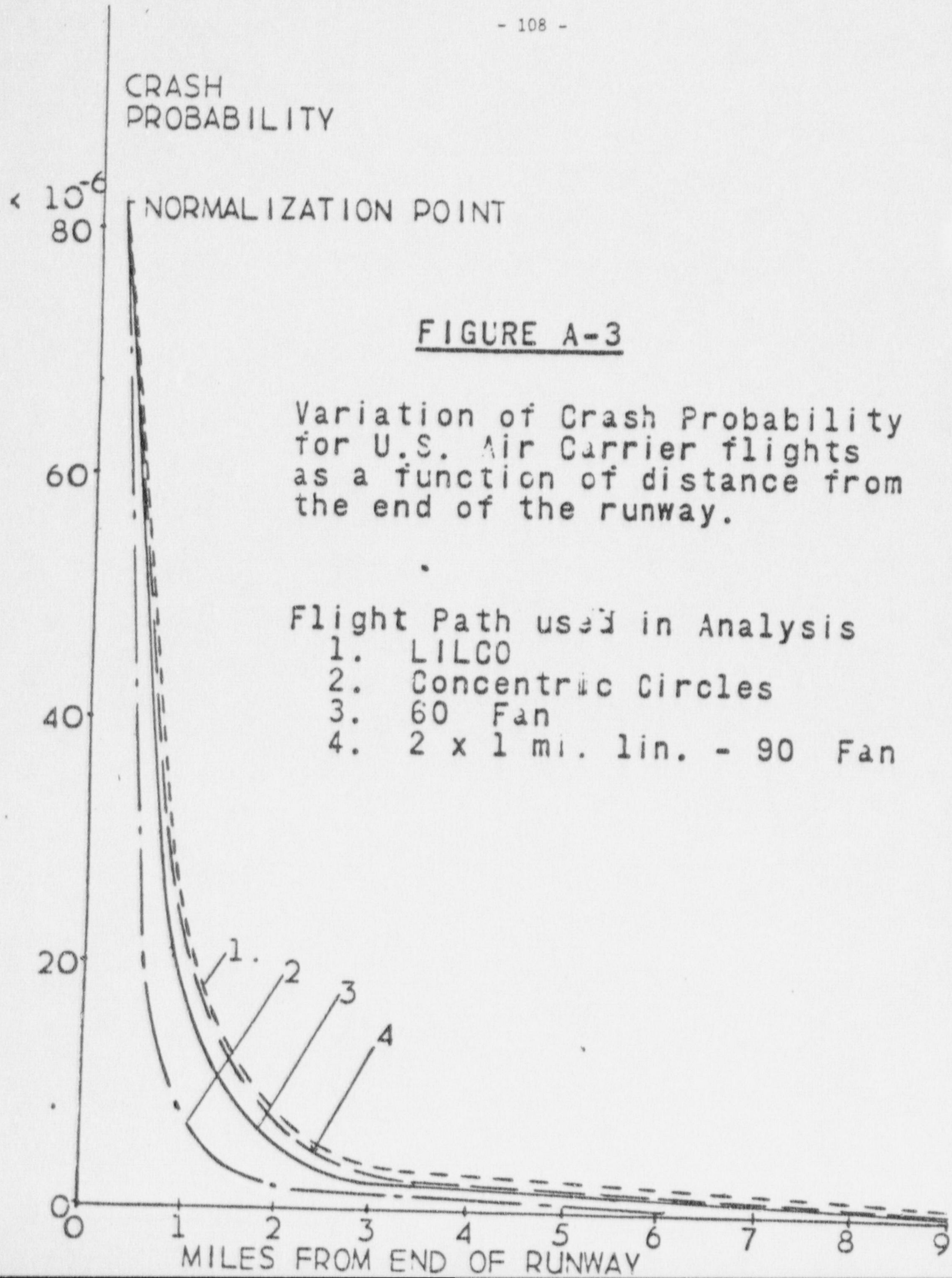

FIGURE A-3

Variation of Crash Probability for U.S. Air Carrier flights as a function of distance from the end of the runway.

Flight Path used in Analysis
1. LILCO
2. Concentric Circles
3. 60 Fan
4. 2 x 1 mi. lin. - 90 Fan

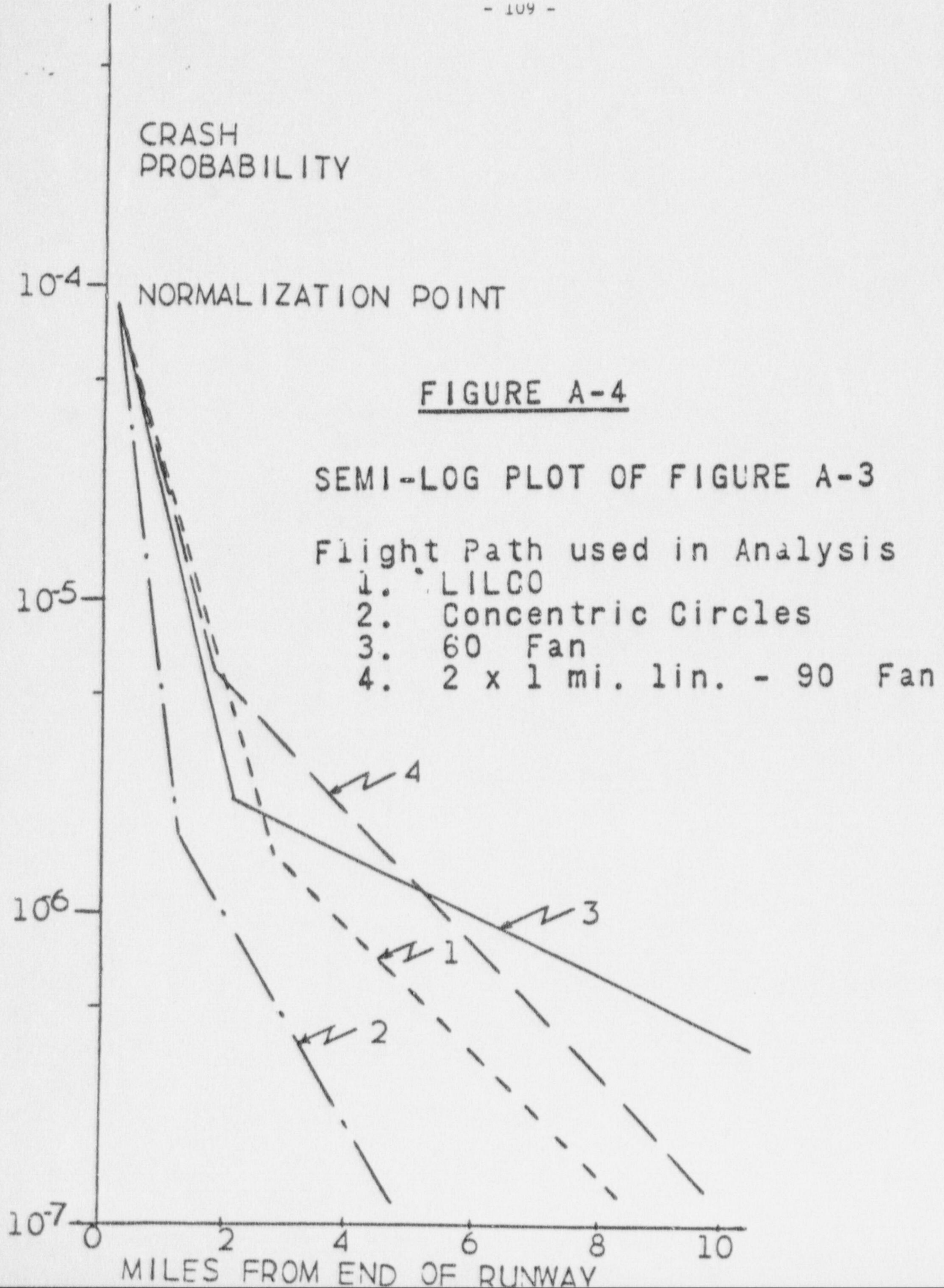

FIGURE A-4

SEMI-LOG PLOT OF FIGURE A-3

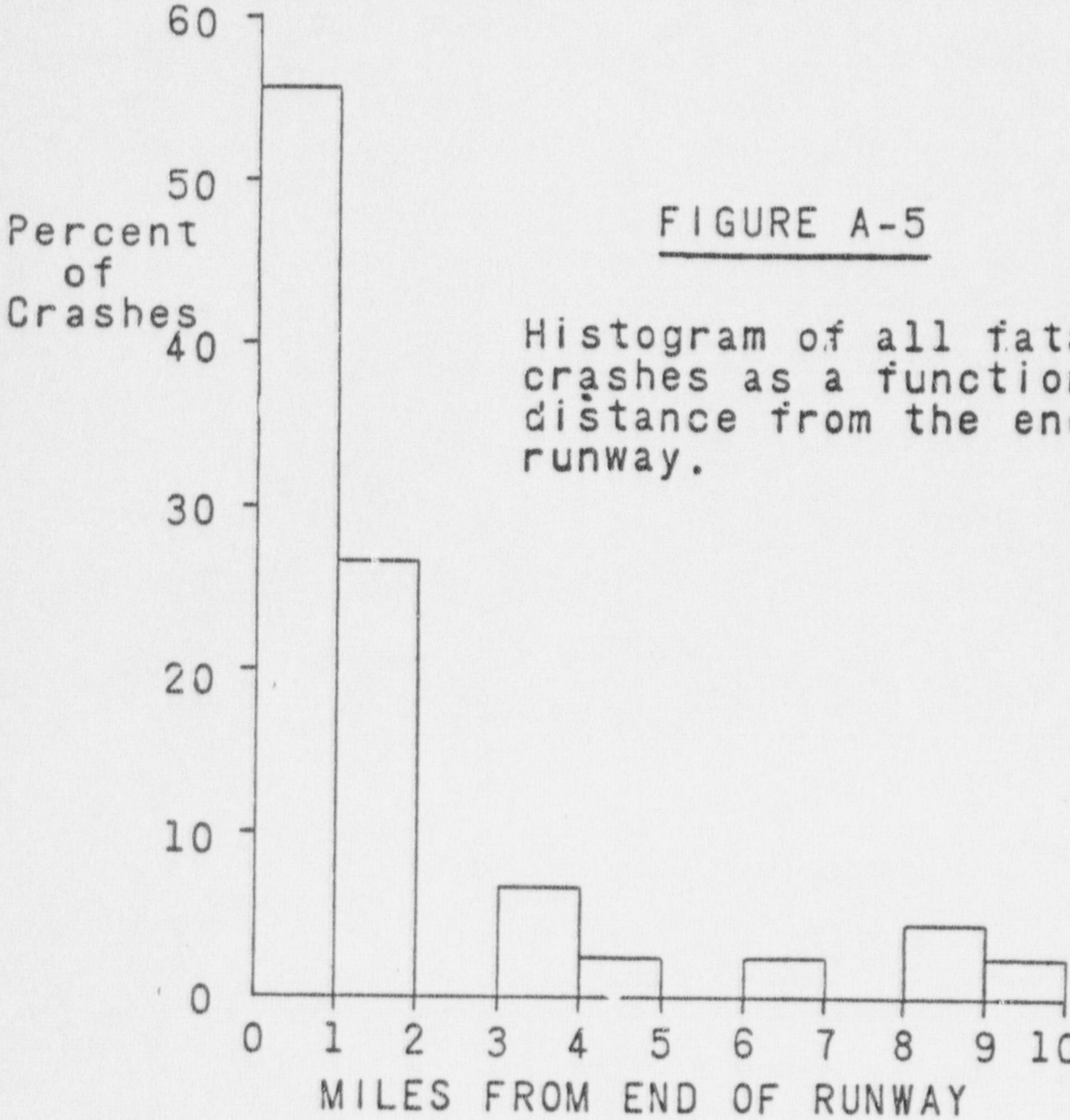

FIGURE A-5

Histogram of all fatal crashes as a function of distance from the end of runway.

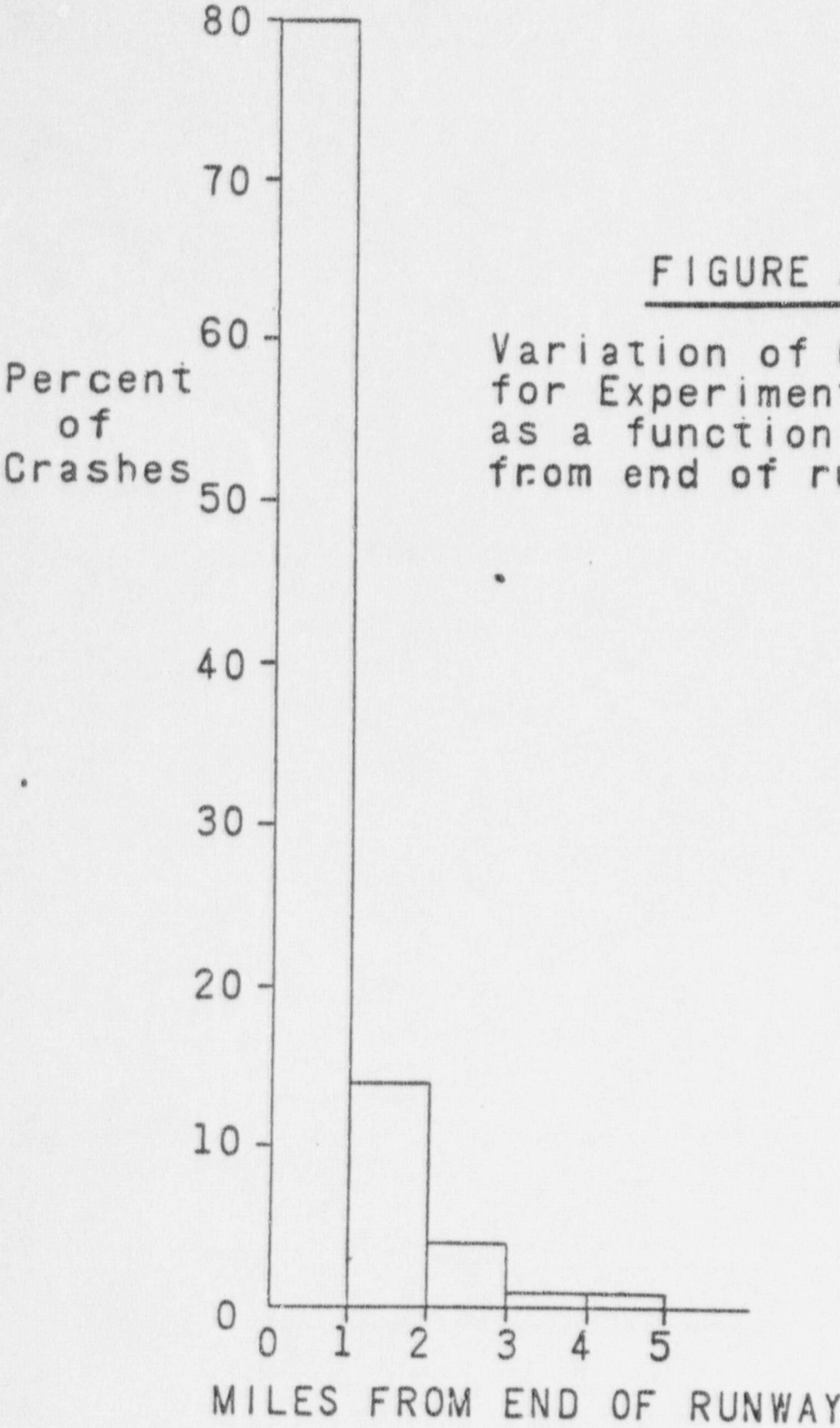

FIGURE A-6

Variation of Crash Density for Experimental Flights as a function of distance from end of runway.

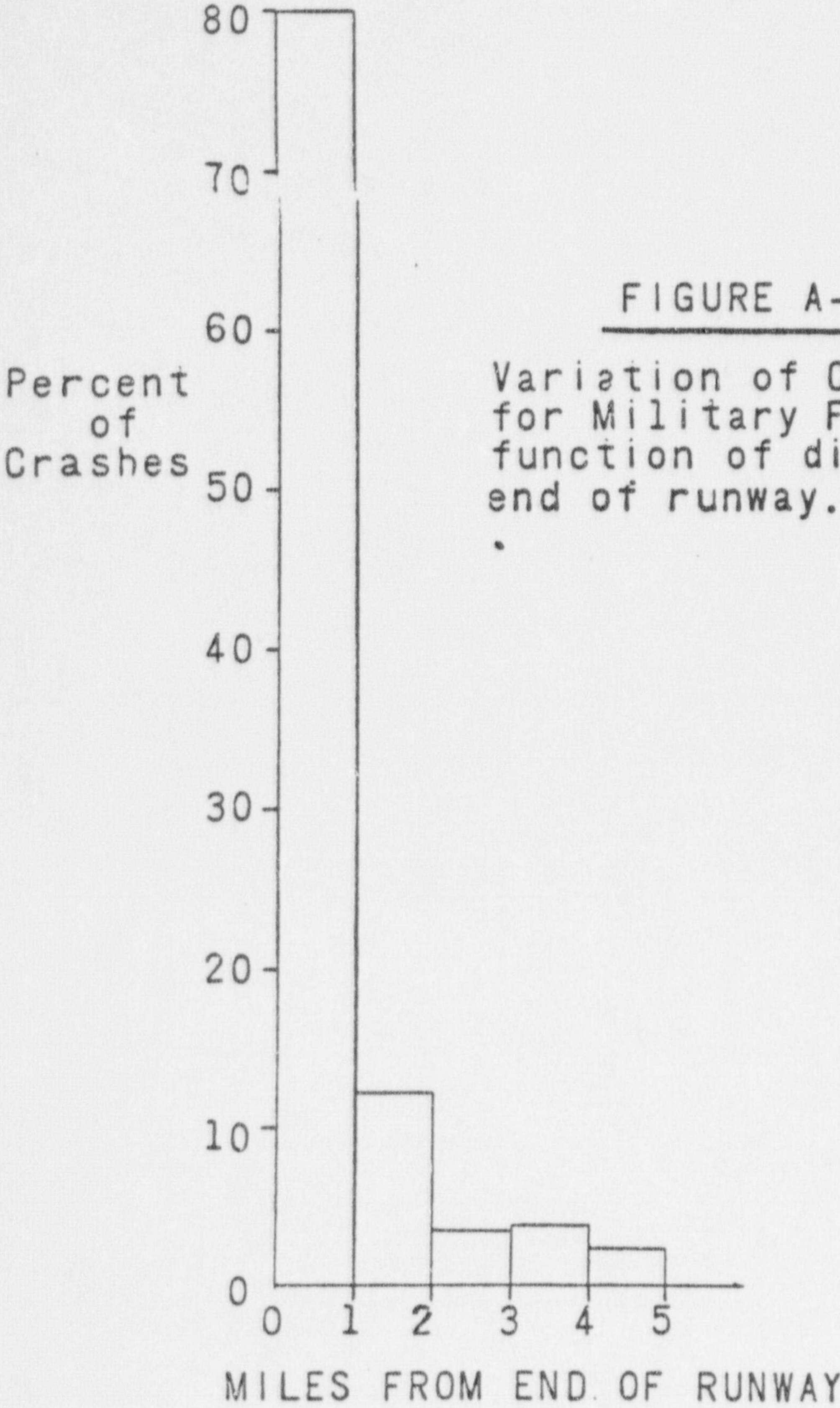

FIGURE A-7

Variation of Crash Density for Military Flights as a function of distance from end of runway.

APPENDIX B

ASSUMPTIONS USED BY THE STAFF IN ACCIDENT ANALYSES

1. Control Rod Drop Accident

    1. The accident occurs 30 minutes after shutdown from full power, (hot standby = worst condition).
    2. Three hundred and thirty fuel rods are failed (based on applicant's analysis).
    3. Peaking factor of 1.5 for failed rods.
    4. 100% of the noble gases and 50% of the iodine fission products in the damaged rods are released.
    5. A decontamination factor of 10 for the iodine passing through the primary system water.
    6. An iodine plateout factor of 2 in the turbine and condenser.
    7. High radiation is sensed in the steamline, signaling the mechanical vacuum pump to stop, and its isolation valves to close so that all radioactivity is contained by the turbine and condenser.
    8. A total leakage rate of 0.5%/day from the condenser, turbine, and turbine building.
    9. The total accident duration is 24 hours.
    10. Ground level release with building wake effect.
    11. Diffusion meteorology discussed in Section 2.2

2. Fuel Handling Accident

    1. 111 fuel rods are failed (based on applicants analysis, equivalent to more than 2 assemblies).
    2. All gap activity in failed rods is released, which is assumed to be 20% of the total noble gases and 10% of the total iodine in the rods.
    3. Water decontamination factor of 10.
    4. Peaking factor of 1.5 for failed rods.
    5. The accident occurs 60 hours after shutdown, (applicant's commitment).
    6. 90% iodine removal by charcoal filters in standby vent system.
    7. Ground level release with building wake effect.
    8. Diffusion meteorology as discussed in Section 2.2.

3. Steam Line Break

    1. Break occurs at full power.
    2. Steam line isolation valves close in 5.5 seconds, (applicant's commitment).
    3. Total mass of coolant released - 16,000 lbs of steam and 45,000 lbs of liquid water.
    4. Release of all iodine and noble gases from the released coolant occurs within two hours.
    5. Coolant activity based on release rate of 0.5 Ci/sec after 30 minutes decay. (Tech Spec limit).
    6. Ground level release with building wake effect.
    7. Diffusion meteorology discussed in Section 2.2.

4. Reactor Coolant System Pipe Break (LOCA)

    1. TID-14844 fission product release (100% noble gases, 25% iodines and 1% solids).
    2. Containment design leakage rate, 0.635% per day for 30 days.
    3. Building mixing credit as discussed in Section 5.6.
    4. Primary containment leakage passes through the standby ventilation system charcoal filter with ar efficiency of 90% for iodine.
    5. Standard man breathing rates.
    6. No correction for plume decay or depletion in transit.
    7. Radioactive decay is accounted for during holdup in the containment.
    8. Ground level release with building effect.
    9. Diffusion meteorology discussed in Section 2.2.

5. Gas Decay Tank Rupture

    1. Release of entire contents of one gas decay tank.
    2. Six hour filling time with 5 cfm turbine in-leakage (worst case).
    3. Inventory in the tank based on 0.5 Ci/sec after 30 minute decay.
    4. Ground level release with building wake effect.
    5. Diffusion meteorology discussed in Section 2.2.