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Shoreham 5% PRA
R. Caruso

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PROBABILISTIC RISK ASSESSMENT
SHOREHAM NUCLEAR POWER STATION
LOW POWER OPERATION
UP TO 5% OF FULL POWER

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For

mobile D/G's, p. 39
- note 2 out of 4 success assumption.

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LONG ISLAND LIGHTING COMPANY
NUCLEAR ENGINEERING DEPARTMENT
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DRAFT

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ABSTRACT

This report assesses the postulated accident sequences which could occur during Shoreham plant operation at low power. The bases for the evaluation are the plant analysis and logic models developed in the Shoreham Probabilistic Risk Assessment. As a measure of public safety, the core vulnerable frequency associated with start-up testing at Shoreham when the power level is restricted to a maximum of 5% of full power is calculated. For this calculation, several changes are made to the plant configuration to reflect the plant as it is, or perceived to be, during the start-up test phase. These changes include:

1) An assumption of no credit for the installed diesel generators;

2) The incorporation of increased AC power system reliability due to the availability of a 20 MW on-site gas turbine with black start capability and the addition of mobile diesel generators to be connected directly to the normal station busses;

3) The incorporation of a detailed procedure for implementing a backup emergency core cooling mode not dependent on AC power in the long term.

One principal focus of the evaluation is on the loss of offsite power (LOSP) initiated accident sequences, based on the judgement that these sequences could be the most important given the assumed start-up configuration. However, all the identified accident sequences from the Shoreham PRA are reexamined to assess the total impact of low power operation. In addition, the requantification is compared with the original Shoreham PRA so that the "risk" associated with start-up testing can be compared on a relative basis with that for normal plant operation.

The conclusion of the analysis is that the core vulnerable frequency for low power operation is much less than that quantified for full power operation, even when quantified conservatively (i.e., taking no credit for installed diesel generator reliability and assuming extended steady state operation at 5% power).

LIST OF ABBREVIATIONS AND ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ASLB	Atomic Safety and Licensing Board
ATWS	Anticipated Transient(s) Without Scram
BWR	Boiling Water Reactor
CET	Containment Event Tree
CCDF	Complementary Cumulative Distribution Function
CHF	Critical Heat Flux
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
D	Demand
DBA	Design Basis Accident
DC	Direct Current
DF	Decontamination Factor
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EHC	Electro Hydraulic Control
EPRI	Electric Power Research Institute
EPS	Electric Power Safeguard
ESF	Engineered Safety Feature
ESW	Emergency Service Water
ET/FTA	Event Tree/Fault Tree Analysis
FSAR	Final Safety Analysis Report
FTA	Fault Tree Analysis
FT/ETA	Fault Tree/Event Tree Analysis
FW	Feedwater
GE	General Electric Company
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
HVAC	Heating Ventilating and Air Conditioning
HX	Heat Exchanger
IBV	Inboard Isolation Valve

LIST OF ABBREVIATIONS AND ACRONYMS (Cont.)

I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronic Engineers
IORV	Inadvertent Open Relief Valve
IREP	Interim Reliability Evaluation Porgram
IRM	Intermediate Range Monitor
LCO	Limiting Condition For Operation
LER	Licensee Event Report
LILCO	Long Island Lighting Company
LIS	Level Indicating Switch
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection (a Mode of RHR)
LPCS	Low Pressure Core Spray (or Core Spray)
LPRM	Local Power Range Monitor
MARCH	Meltdown Accident Response Characteristics
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
MTTR	Mean Time To Repair
NRC	Nuclear Regulatory Commission
NSS	Normal Station Service
NSSS	Nuclear Steam Supply System
OBV	Outboard Isolation Valve
PCS	Power Conversion System
P&ID	Piping and Instrumentation Drawing
PRA	Probabilistic Risk Assessment
PRM	Power Range Monitor
PRS	Pressure Relief System
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RPS	Reactor Pressure Vessel

LIST OF ABBREVIATIONS AND ACRONYMS (Cont.)

RPT	Recirculation Pump Trip
RSS	Reactor Safety Study
RWCU	Reactor Water Clean-up
SAR	Safety Analysis Report
SDV	Scram Discharge Volume
SF	Shielding Factor
SFSP	Spent Fuel Storage Pool
SGTS	Standby Gas Treatment System
SIA	Service and Instrument Air System
SJAE	Steam Jet Air Injector
SLC	Standby Liquid Control
SNPS	Shoreham Nuclear Power Station
SORV	Stuck Open Relief Valve
SP	Suppression Pool
SPC	Suppression Pool Cooling
SRM	Source Range Monitor
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
SVERM	Station Ventilation Exhaust Radiation Monitoring
SW	Service Water
SWS	Service Water System
TBCLCW	Turbine Building Closed Loop Cooling Water System
TCV	Turbine Control Valve
TG	Turbine Generator
TIP	Traversing In-Core Probe
UHS	Ultimate Heat Sink
UPS	Uninterruptiable Power Supply

1.0 INTRODUCTION

The Shoreham Nuclear Power Station may pose a small incremental risk to the public. The Shoreham Probabilistic Risk Assessment [1-1] performed for LILCO calculated this risk (i.e., the product of probability and potential consequences) in a manner consistent with the current state of the technology. The results indicate that the core melt frequency and the risk measures of the proposed safety goals are met. The PRA was performed assuming plant operation in the normal power mode, i.e., approximately 100% full power. The purpose of this report is to quantify one of the key parameters derived in the PRA-the frequency of core vulnerable conditions-associated with plant operation at or below 5% of full power. Other risk parameters, e.g., source term characteristics, will also be discussed to provide additional qualitative information and scaling factors to assess the relative risk of the low power operation.

The principal impact of this evaluation should be in the form of a reasonableness test on the relative plant risk associated with operation of the plant during start-up testing versus normal mature plant operation. There are both positive and negative effects which may influence the relative risk associated with low power operation. These effects are factored into this assessment. The results of the probabilistic logic model quantification are one input into the decision-making process. However, the principal test which the results are intended to provide the decision maker is the reasonableness of low power operation. Specifically, an evaluation of the relative risk contribution can provide insights as to whether low power start-up testing represents a disproportionate contribution to risk.

1.1 SCOPE

As stated above, the objective of this report is to quantify the probabilistic models of the Shoreham plant in order to assess the frequency of core vulnerable conditions when operating at 5% of full power. This quantification is based upon the PRA logic models and uses many of the ground rules established there. The analysis given here is directly consistent with the PRA quantification and therefore affords a reasonable relative "risk" comparison between full power

operation and the restricted power level case. This power level restriction has a number of positive benefits which tend to reduce risk compared to operation of a mature plant at 100% power:

- The lower initial power level and lower decay heat levels reduce the rate of coolant inventory loss and the heatup of containment, thereby increasing the time available for operator action.

- The reduced requirements for coolant makeup and containment heat removal allow greater flexibility in mitigating accidents, changing the plant system success criteria (e.g., the viability of CRD flow as a successful coolant injection path for non-ATWS non-LOCA scenarios).

- In the unlikely possibility of a core vulnerable condition there is an increased time available for emergency response.

In addition, the following positive considerations have been factored into the evaluation of low power operation at Shoreham:

- Response capability of on-site portable generators with special connections to the normal power buses.

- Response capability of on-site blackstart gas turbine.

- Potential administrative controls which require shutdown in the face of severe weather, e.g., high winds, hurricane, tornado watch, etc.

- Increased time available for preservation of containment integrity.

- Reduction in radionuclide core inventory due to the early in life conditions of the fuel.

Negative effects associated with the start-up operations and the 5% power limitation include:

- o The initial period of plant operation tends to exhibit a substantially higher transient challenge frequency and a higher system unreliability than a mature plant.
- o The testing phase of the plant power ascension may introduce even higher transient frequencies.
- o With more control rod motion there may be a possibility for unusual power shapes, and control rod withdrawal incidents. This has been explicitly addressed in the modeling of transient phenomena presented in Appendix B.

1.2 LIMITATIONS

A key point to consider is that the Shoreham PRA does not include analysis of external events. One reason for not yet considering external events is the large uncertainty associated with the external event evaluation compared to other accident scenarios associated with full power operation. Nevertheless, external events may contribute to the potential risk associated with low power operation+. By recognizing this limitation, the quantified Shoreham PRA can be effectively used in a risk based discussion or low power operation. The PRA is a valuable tool to be used in setting priorities and as one input to the decision-making process.

1.3 DESCRIPTION OF LOW POWER OPERATION

The overall objective of low power operation at Shoreham is to gain operating experience. This applies to both LILCO personnel and plant systems. As such, plant safety systems, and alternate plant systems that may play a significant safety role for accident mitigation during sequences initiated at low power. many aspects of low power operation are significantly different from operation of a mature plant at 100% power. The purpose of this discussion is to outline

+Effects of external events are anticipated to be small based upon commitments made by LILCO regarding limiting conditions of operation [1-2].

these operational differences and to define the plant configuration that is assumed in this analysis. This includes the operation of normal plant systems,

1.3.1 General Plant Configuration

Plant operation at a maximum of 5% power is assumed to involve one leg of the feedwater/condensate system for RPV level control, steam flow through all four main steam lines, and bypass steam flow directly to the main condenser. Reactor recirculation flow will be controlled by having both recirculation pumps in operation.

All safety systems used for scram, RPV pressure control, RPV coolant injection, and containment control are assumed to be available within the technical specification limits that apply to full power operation. Safety systems that perform support functions are also assumed to be available according to technical specifications, with special consideration in one case: on-site emergency electric power systems. This analysis provides estimates of system availability assuming the installed emergency diesel generators are unavailable, which means that essentially no credit is taken for the installed diesel generators.

Emergency power supply restoration capability following loss of off-site power will be augmented by the presence of an on-site 20 MW gas turbine with auto-start (blackstart) capability, and the presence of mobile generator units which can be connected to the plant side of the RSS transformer, both of which are included explicitly in this assessment. Low power operation offers the safety advantage of reduced decay heat levels following a scram. Figure 1.3-1 provides simplified schematic of the range of operation under consideration, i.e., from approximately 1% to 5% of full power. Deterministic calculations form the basis for assuming that several additional plant contingencies (many of which were considered for completeness, but given very little credit in the containment event trees of the PRA) may be available in emergencies. Two methods for RPV injection which are judged viable alternatives for some accidents associated with low power operation include: (1) the CRD hydraulic system pumps (one of which is in continuous operation), is found adequate due to the reduced coolant makeup requirements; and, (2) fire water injection through a spoolpiece

connection to the ultimate core cooling connection has increased probability of success due to the extended times available for implementation. Both alternatives are results of the lower decay heat levels associated with power operation at less than 5% power.

1.3.2 Specific Aspects of Low Power Operation

There are many additional factors involved in the safety evaluation of operation at or below 5% power. The following considerations have been included in the analysis: [1-2]

- LILCO will implement administrative controls which require shutdown in the face of severe weather (e.g., high winds, hurricane, tornado watch, etc.)

  This will have a favorable impact on the evaluation of offsite power reliability, allowing failures due to severe weather to be eliminated from the data base.

- The time available for effective operator action to restore or repair equipment (e.g., restoration of offsite power) is increased due to the lower reactor power.

  This is primarily found to have a favorable impact on the evaluation of AC power system reliability since the additional time available to the operator increases the probability of successful actions.

- The time available for preservation of containment integrity is increased.

  This has a dramatic impact on accident sequences involving challenges to containment or containment-related systems interactions phenomena (e.g., HPCI/RCIC failure due to high lube oil temperature). For the most part, this impact is reflected in the revised success criteria.

- o Early in life (referred to as "wear-in") failures (including initiating events and/or equipment failures) have been shown by operating experience to be more likely than for operation of a mature plant.

  This is reflected in the analysis through increases in estimates of system unavailabilities and initiating event frequencies.

1.4 SUCCESS CRITERIA

In this report, best estimate calculations of core vulnerable frequency are made assuming that the equilibrium power level is approximately 5%. An estimate of the sensitivity of core vulnerable frequency to this assumption is also derived based on an equilibrium power level value of 2.5% as discussed in Appendix E.

The system success criteria for low power operation are derived from the Shoreham PRA with the exception that there are additional success paths which do not exist for cases in which the plant operates at high power. Tables 1-1 and 1-2 provide the system success criteria for low power operation in a format similar to the Shoreham PRA.

The principal examples of the additional system success states for low power operation are as follows:

- o For transients with no SORVs, flow from the CRD pumps, the diesel fire pump or any of the service water pumps would provide adequate coolant injection.

- o For SORV or medium LOCA cases, no additional depressurization systems are required to allow low pressure systems to inject.

- o For ATWS conditions, RCIC plus CRD flow would provide adequate coolant injection for powers in the range of 2.5%.

1.5 REPORT ORGANIZATION

Section 2.0 of this report provides a summary discussion of the Shoreham PRA accident sequence analysis. This includes a review of the dominant contributors to the frequency of core vulnerable conditions. The sequences defined in Section 2.0 are reviewed in Section 3.0 to assess their impact for operation at 5% power. For convenience, four categories of sequences are discussed: Loss of offsite power induced transients, LOCAs, other transients, and ATWS sequences. Sections 4.0 and 5.0 provide a summary of the low power operation sequence quantification results and conclusions of the analysis, respectively.

FIGURE 1.3-1

RANGES OF NEUTRON MONITORING SYSTEM

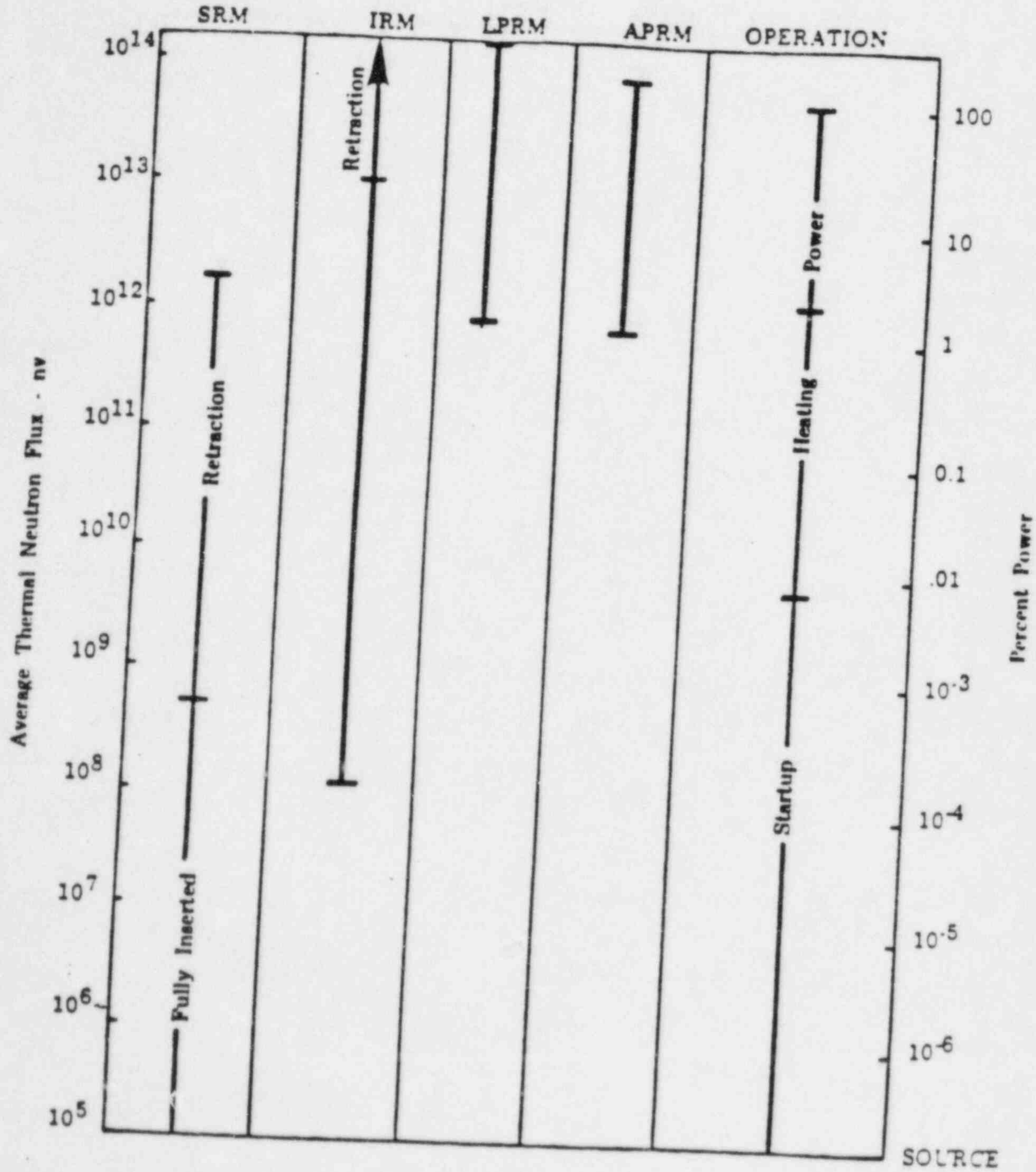

TABLE 1.1
SUMMARY OF SUCCESS CRITERIA FOR THE MITIGATING SYSTEMS TABULATED AS A FUNCTION OF ACCIDENT INITIATORS

ACCIDENT INITIATOR	SUCCESS CRITERIA	
	COOLANT INJECTION	CONTAINMENT HEAT REMOVAL
Large LOCA: Steam Break 0.08 ft^2 Liquid Break 0.1 ft^2	1 of 4 LPCI Pumps Or 1 of 2 Core Spray Pumps Or Diesel Fire Pump Or 1 of 4 Service Water Pumps	1 RHR
Medium LOCA: Steam Break 0.016 to 0.08 ft^2 Liquid Break: 0.004 to 0.1 ft^2	Same as Large LOCA	1 RHR
Small LOCA: Steam Break 0.016 ft^2 Liquid Break 0.004 ft^2	CRD Or HPCI Or RCIC Or 1 Feedwater Pump Or ADS* and 1 of 2 CS Pumps Or 1 of 4 LPCI Pumps Or 1 Condensate Pump Or Diesel Fire Pump Or 1 of 4 Service Water Pumps	Normal Heat Removal Or 1 RHR Or RCIC in St. Cond. Mode
Transient	Same as Small LOCA	Same as Small LOCA
IORV	Same as Large LOCA	Same as Large LOCA
Transient + SORV	Same as Large LOCA	Same as Large LOCA

*ADS refers to any mode of successful reactor depressurization.

TABLE 1-2

SUCCESS CRITERIA FOR ATWS SEQUENCES
BASED ON MODIFICATIONS IMPLEMENTED AT SHOREHAM

TRANSIENT INITIATING EVENT	EFFECT OF POTENTIAL ADDITIONAL FAILURES (in Addition to ARI Failure)[a]							
	REDUCED OR LATE POISON PROJECTION	REDUCED COOLANT INJECTION		REDUCED SUPPRESSION POOL COOLING		PRESSURE RELIEF	OTHER ATWS FEATURES	
		FW	FW & HPCI	1 RHR	BOTH RHRs		RPT	ADS INITIATED
MSIV CLOSURE	A	A	A	N	N	N	A	A
TURBINE[b] Trip	A	A	A	A	N	N	A	A
IORV	A	A	A	N	N	N	A	A
LOS OF OFFSITE POWER	A	A	N	N	N	N	A	A
LOSS OF FEEDWATER	A	A	A	A	N	N	A	A
LOST OF CONDENSER	A	A	A	N	N	N	A	A

A = Acceptable (Successful): acceptable implies no significant fuel damage and suppression pool temperatures less than 240°F

N = Not Acceptable (Not Successful)

(a) = Combinations of failures not shown on the above table as acceptable should be considered unacceptable. These success criteria can be used to evaluate the successful states of the plant following an ATWS from less 25% power. Note that RPT is not required for sequences from 25% power or less.

(b) = All changes in recirculation flow outside of acceptable limits are treated as leading to a turbine trip as are all increasing feedwater flow transients

REFERENCES

1-1 Probabilistic Risk Assessment, Shoreham Nuclear Power Station, Long Island Lighting Company, Docket 50-322, June 1983

1-2 Supplemental Motion for Low Power Operating License, submitted in the matter of Long Island Lighting Company, Docket No. 50-322, to the Atomic Safety and Licensing Board, affidavit of W. J. Museler, dated March 20, 1984.

2.0 PRA SUMMARY

The purpose of this section is to briefly summarize the quantitative results of the PRA which can then be used for reference throughout the remainder of this report. Many of the technical appendices of the Shoreham PRA provide information used for the quantification of the probabilistic models. Except for the differences explicitly described in this analysis, essentially all of the input data are the same; thus forming the basis of this comparative analysis.

2.1 ACCIDENT CLASSES

Because the PRA is intended to provide a quantitative measurement of risk to the public, several factors which influence the risk calculations other than the frequency of releases are considered. For this reason, five accident classes are chosen which are intended to represent the spectrum of accidents from (relatively) high frequency/low consequence events to low frequency/high consequence events. These accident classes are defined as follows:

Consequence	CLASS	DESCRIPTION
low	I	Inadequate Coolant Inventory Makeup
	II	Inadequate Decay Heat Removal
	III	LOCA With Inadequate Coolant Inventory Makeup
	IV	ATWS with Inadequate Containment Heat Removal
high	V	Interfacing LOCA

A second aspect of the analysis is the evaluation of various initiating events. The PRA attempts to quantify the different impacts on plant response under a variety of initial conditions, ranging from the more common types of anticipated transients to very low frequency initiators. Together, these two aspects of the probabilistic analysis-initiating events and accident classes-are summarized in Table 2-1.

TABLE 2-1

SUMMARY OF THE DOMINANT ACCIDENT SEQUENCE FREQUENCIES WHICH LEAD TO CORE VULNERABLE STATES (PER REACTOR YEAR) BY INITIATOR AND CLASS

EVENT INITIATOR CLASS	CLASS I	CLASS II	CLASS III	CLASS IV	CLASS V	SEQUENCE* TOTALS
Transients:						
Turbine Trip	2.5E-6	1.0E-6	--	--	--	3.5E-6
Manual Shutdown	1.4E-6	1.2E-6	--	--	--	2.5E-6
MSIV Closure	7.4E-7	3.5E-7	--	--	--	1.1E-6
Loss of Feedwater	2.0E-7	4.2E-8	--	--	--	2.4E-7
Loss of Condenser Vacuum	3.2E-6	2.1E-6	--	--	--	5.2E-6
Loss of Offsite Power	9.9E-6	5.7E-7	--	--	--	1.0E-5
IORV	6.8E-7	8.9E-8	--	--	--	7.7E-7
	1.7E-5	5.9E-6				2.4E-5
LOCA:						
Large LOCA	--	6.9E-7	1.8E-7		--	8.7E-7
Medium LOCA	--	2.7E-7	5.1E-7	3.0E-8	--	8.0E-7
Small LOCA	2.1E-7	2.8E-8	1.5E-8	--	--	2.6E-7
LOCA Outside Containment	--	7.2E-9	--	--	3.6E-8	4.3E-8
Reactor Pressure Vessel LOCA	--	--	3.1E-7	--	--	3.1E-7
	2.1E-7	9.9E-7	1.0E-6	3.7E-8	3.6E-8	2.3E-6
ATWS:						
Turbine Trip	1.2E-6	--	8.5E-10	2.3E-6	--	3.5E-6
MSIV Closure/Loss of Condenser Vacuum	8.0E-7	--	7.5E-10	7.4E-6	--	8.2E-6
Loss of Offsite Power	7.1E-8	--	--	6.9E-7	--	7.6E-7
IORV	1.7E-7	--	--	1.6E-7	--	3.3E-7
Loss of FW	1.8E-6	--	2.1E-9	3.0E-6	--	4.8E-6
	4.0E-6		3.7E-9	1.4E-5		1.8E-5

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EVENT INITIATOR CLASS	CLASS I	CLASS II	
Other Transients: Cases Involving the Release of Excessive Water	3.1E-6	7.8E-	
Cases Initiated by the Loss of DC Power Bus	2.7E-6	7.4E-8	
Cases Involving an Upset Condition with the Reactor Water Level Measurement System	2.4E-6	1.2E-7	
Manual Shutdown Due to High Drywell Temperature	1.4E-7	--	--
	2.5E-6	1.2E-7	
Loss of Service Water Initiated Events	3.1E-7	6.9E-7	
TOTAL	3.2E-5	8.0E-6	1.0E-6

*Note: Totals may not match due to round off errors.
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The plant configuration and the unique mitigation features available at the restricted 5% power level are judged to impact many of the dominant accident sequences in a similar manner. Many of the initiators described in the PRA can be combined in this analysis for the purposes of subsequent discussion. Specifically, the initiator types are discussed in the following categories since the power restriction of 5% has a similar impact on the resulting postulated accident sequences;

1. Loss of Offsite Power
2. Loss of Coolant Accidents
3. Other Transients
4. ATWS

Table 2-2 is a reduced version of Table 2-1 presented in terms of these four generalized types of postulated scenario initiators. The quantification and discussion in Section 3 is divided according to these generalized "initiator types". The essence of this slight change in focus results from the following considerations:

- Long term containment heat removal (Class II) accident sequences which are calculated to be of low frequency in the Shoreham PRA are found to be significantly smaller in frequency for the start-up testing mode of operation at Shoreham.

- The interfacing LOCA sequences (Class V) are very low frequency sequences and not a dominant contributor to the core vulnerable frequency.

- The character of the LOSP sequences and the interest in the sensitivity of the results to installed diesel generator reliability has elevated their potential importance during the low power startup phase.

TABLE 2-2

DOMINANT ACCIDENT SEQUENCE FREQUENCIES AT 100% POWER
(PER REACTOR YEAR) BY INITIATOR TYPE

Event Initiator Type	CLASS I	CLASS II	CLASS III	CLASS IV	CLASS V	SEQUENCE TOTALS
Loss of Offsite Power	9.9E-6	5.7E-7	--	--	--	1.0E-5
LOCAS	2.1E-7	9.9E-7	1.0E-6	3.7E-8	3.6E-8	2.3E-6
Other Transients	1.8E-5	6.4E-6	--	2.8E-8	--	2.4E-5
ATWS	4.0E-6	--	3.7E-9	1.4E-5	--	1.8E-5
TOTAL	3.2E-5	8.0E-6	1.0E-6	1.4E-5	3.6E-8	5.4E-5

3.0 LOW POWER RISK EVALUATION

The focus of this section is to quantify the probabilistic models of the Shoreham plant [3-1] in order to assess the frequency of core vulnerable conditions when operating at the 5% power level. This quantification is based upon the PRA logic models and uses many of the groundrules established there. The analysis given here is directly consistent with the PRA quantification and therefore affords a relative "risk" (i.e., frequency of core vulnerable) comparison between full power operation and the restricted power level cases. The evaluation of core vulnerable conditions is based on a point estimate equilibrium operating value of approximately 5% of full power. Since the intended use of the low power testing is to restrict the power level to below 5%, the anticipated equilibrium power level for characterizing the potential decay heat levels may be closer to an effective 2.5% power due to power cycling for training and testing. However, there may also be some uncertainty in the calibration of the power level. Therefore, it is judged prudent to perform the base calculations assuming a power level of 5% and report the sensitivity of the power level changes to approximately 2.5% power in Appendix E.

Previous NRC investigations (3-2, 3-3) have indicated that, in general, the risk to the public and the frequency of potential core vulnerable conditions are considerably lower for low power operation than the estimates calculated in the Shoreham PRA and other PRAs for normal full power operation. The Shoreham specific analysis performed here documents evaluations similar to those performed by the NRC (3-2, 3-3) on a plant specific basis for Shoreham. The Shoreham PRA is used as the baseline analysis to establish the relative "risk" and core vulnerable frequency for normal power operation. In addition to the conditions which may exist during normal operation, this analysis considers several variations in assessed initial plant configuration. The principal change in the initial plant configuration which is examined is the availability of the installed diesel generators and alternative backup methods for obtaining AC power restoration. Because this change has the strongest effect on the postulated Loss of Offsite Power (LOSP) initiators and since the resulting sequences are among the largest contributors to the frequency of core vulnerable conditions, these sequences are examined separately.

Section 3 addresses the quantification of the four generalized types of initiators:

- o LOSP (Class I) - Section 3.1
- o LOCA (Class I & III) - Section 3.2
- o Transients (Class I & II) - Section 3.3
- o ATWS (Class I & IV) Section - 3.4

3.1 LOSS OF OFF-SITE POWER INITIATOR

The LOSP initiator represents a unique accident challenge since it causes the unavailability of the normal system used to supply coolant makeup and containment heat removal. This section is structured to discuss several possible variations in the scenario and in the plant configuration. As such, it is important to estimate the timing of LOSP sequences with respect to key plant parameters. The following discussion is based on the deterministic results obtained in Appendix B which are also summarized in Section 3.1.2.

The containment conditions during LOSP initiated scenarios from below 5% power will be substantially less severe than the containment conditions calculated for sequences originating from 100% power. As an example, one of the previously identified plant conditions which may contribute to the RCIC and HPCI failure probability is the potentially high suppression pool temperature+ and containment pressure (RCIC)++ which may occur within 7 to 10 hours following a station blackout from full power. The analysis reported in Appendix B shows that such adverse conditions are not expected to occur for many hours beyond these estimates. Thus, the limiting sequence involving plant recovery actions involves a loss of coolant injection following LOSP, unrelated to containment conditions.

+ Failures of HPCI and RCIC lubrication cooling are postulated if suppression pool temperatures are high and pump suction is from the suppresion pool.

++ Very high containment pressures in the range of containment design pressure could result in protective trip actuation on RCIC.

To provide a perspective on the LOSP evaluation, LOSP from high initial reactor power coupled with failures of coolant injection systems are found to present potential core vulnerable conditions within 30 minutes to 1 hour. However, similar cases of LOSP at 5% power are found to present potential core vulnerable conditions only at times greater than 3 hours, and, for many cases, at times greater than 30 hours. From this perspective, it can be seen that the 5% power restriction sharply increases the time available for operating staff actions to establish successful coolant injection and AC power restoration.

A detailed event tree analysis is performed to identify the principal contributors to potential core vulnerable situations, given the following initial conditions:

- o LOSP initiator.
- o Initial power 5.0% for over 30 days.
- o Installed diesel generators assumed unavailable; sensitivity studies have also been performed to provide a more realistic estimate of on-site AC power reliability.
- o On-site, black start gas turbine is available with a reliability developed from machine specific data.
- o Additional temporary on-site AC power enhancements including:
    - portable diesel generators which can be manually switched into the 4160V AC power system in the event of a station blackout.
    - restrictions on reactor power operation in the event of severe weather warnings, e.g., hurricanes or tornado.

Figure 3.1 (a,b,c,d,e,f) is the LOSP event tree used in the evaluation of potential core vulnerable conditions from low power. The event tree is constrained to the delineation of a relatively small number of sequences in order to simplify the presentation. The evaluation of LOSP sequences is an

iterative process because of the time dependence of recovery events. In other words, the probability of a specific sequence depends strongly on the associated sequence timing, while, at the same time sequence timing is calculated for the most probable sequence variations. As a result, Figure 3.1 and the corresponding calculations in Appendix B are developed in parallel. Figure 3.1 consists of a screening event tree (Figure 3.1a) that is used to define 5 groups of sequences each with its own similar timing. Each grouping of sequences is then modeled on a subsequent event tree (Figure 3.1b,c,d,e,f) using the results of Appendix B as a basis for estimating event probabilities. Thus, time dependent effects are not sequentially mapped out on the initial event tree, but are judgementally included in estimating of failure probabilities.

The functional events in the LOSP event tree shown in Figure 3.1a are discussed below.

Initiating Event (T): The quantification of the LOSP initiating frequency is based upon LILCO grid specific data and is consistent with the original Shoreham PRA (see also Appendix A).

Scram (C): The scram system reliability is taken from the Shoreham PRA which in turn used a point estimate value from the NRC (NUREG-0460). The value shown in Figure 3.1 is the mechanical common mode failure probability of the control rods to insert. Note that, consistent with the higher failure rates assumed for the wear-in period of plant operation, this failure probability has been increased by 100%. The electrical common mode failure probability is similar to that in the Shoreham PRA and has been shown to lead to a significantly lower frequency of core vulnerable since LILCO has incorporated a backup electrical scram system (ARI) in the Shoreham design.

Primary System Integrity (P): One of the important parameters in the plant response to a transient from low power is the ability to maintain the primary system integrity. If integrity is maintained, coolant is lost only by intermittant SRV actuation and the calculated time to a core vulnerable condition is extremely long, i.e., greater than 30 hours. However, if the reactor is depressurized through an SORV or LOCA condition, then the time to core uncovery and reactor fuel heatup is substantially less, i.e., approximately

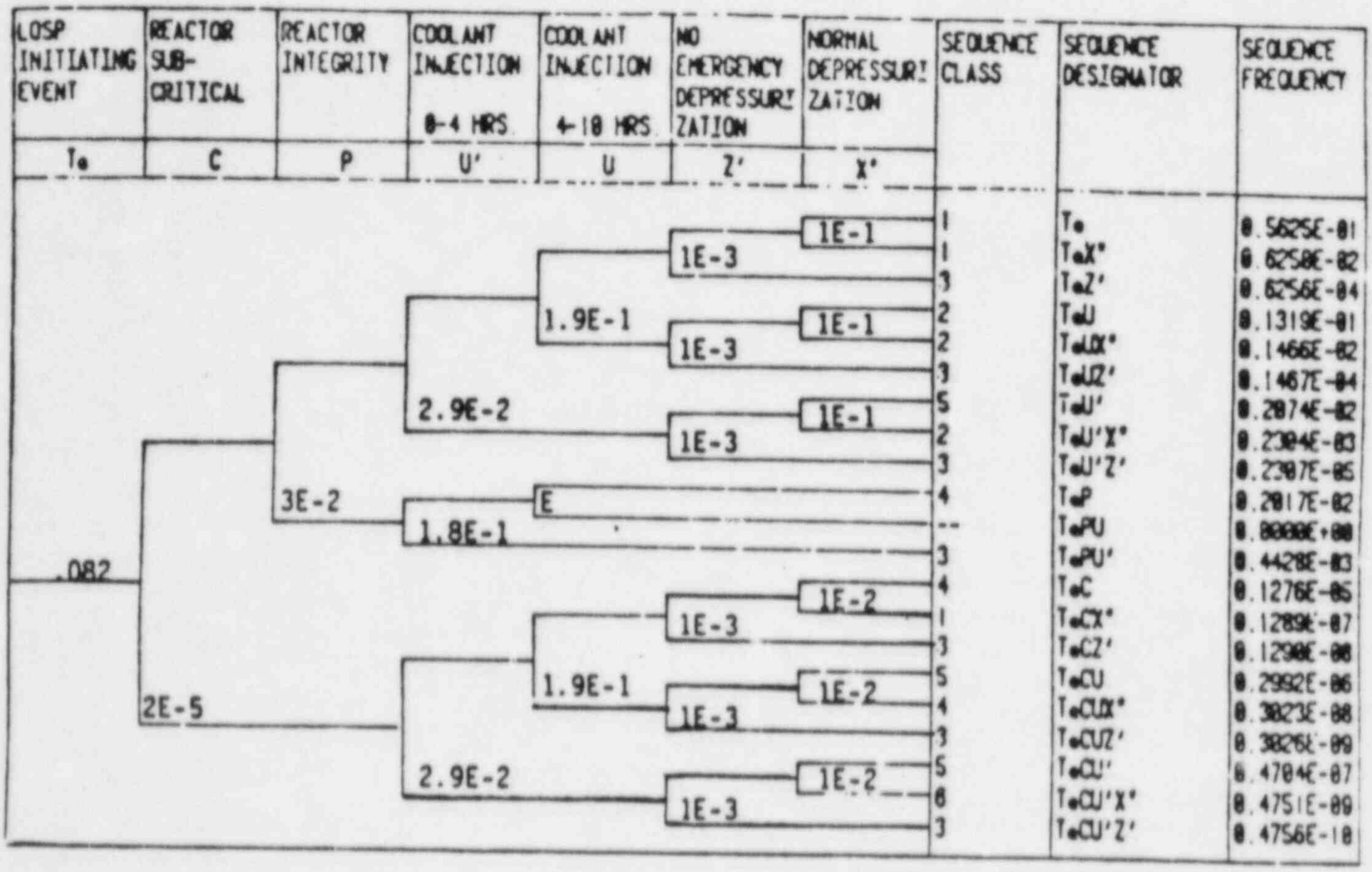
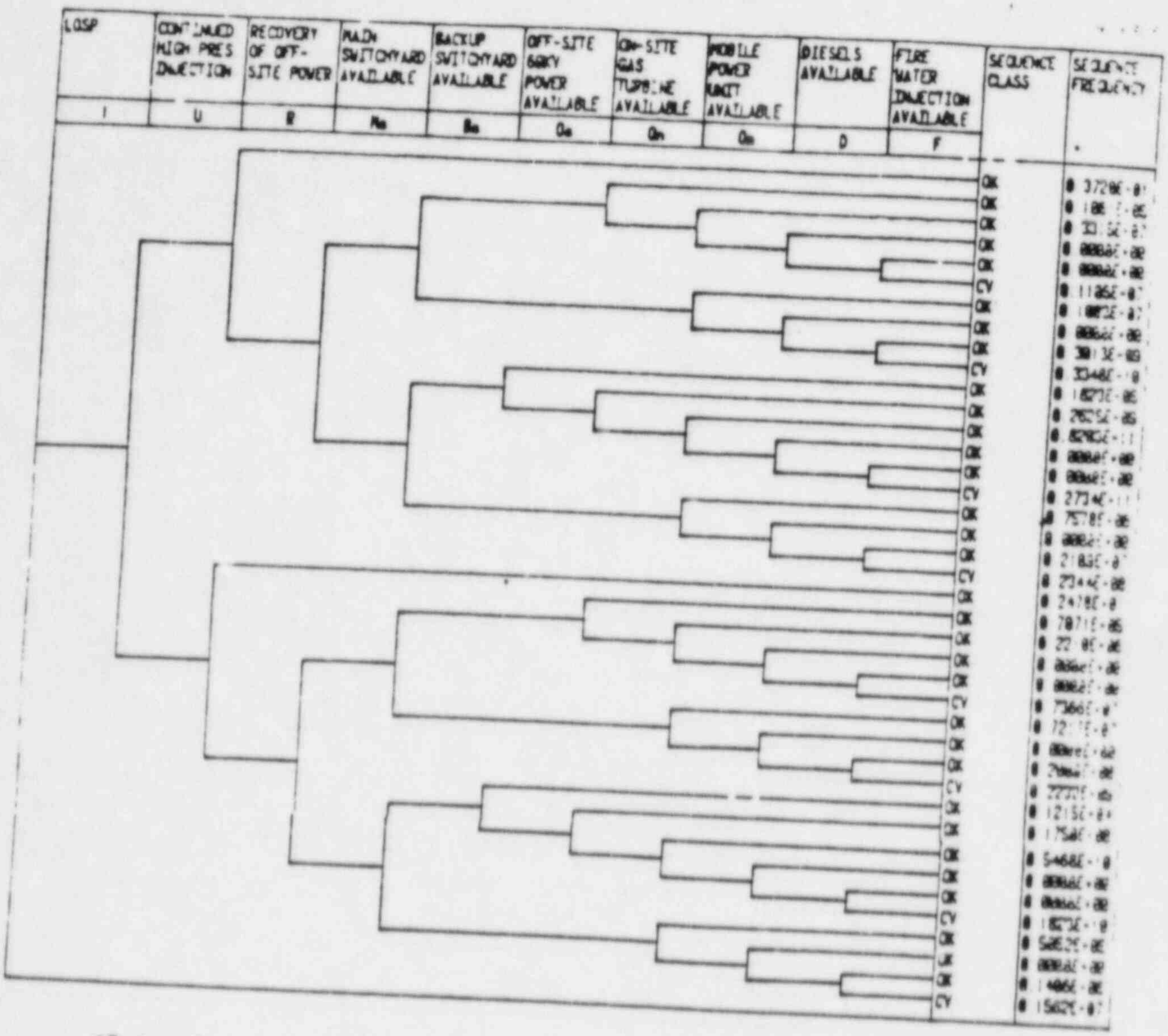


FIGURE 3.1a

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)



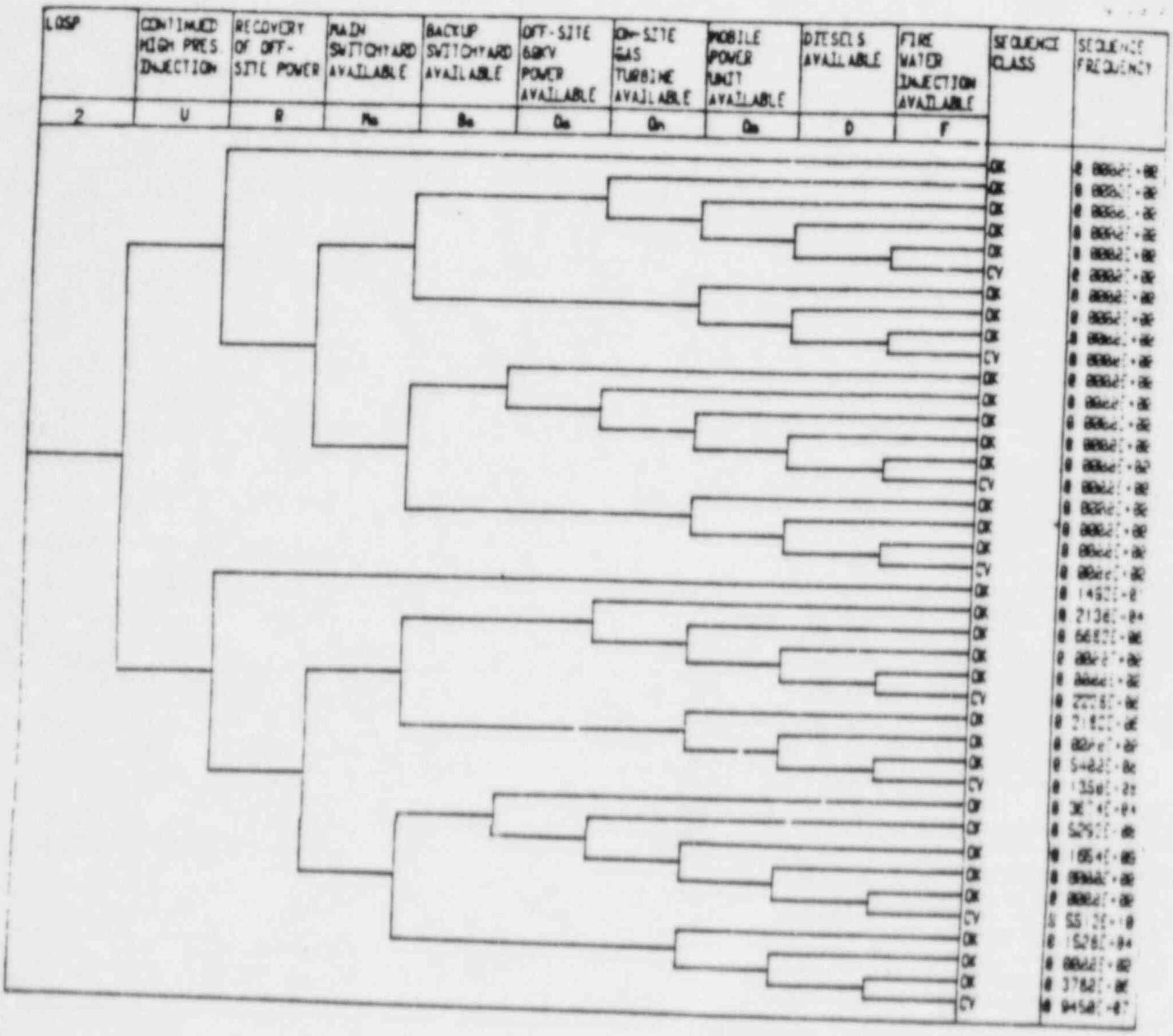
*Entry Conditions Sequence Type 1 : LOSP; Isolation; Reactor Scrammed; Primary System Intact; Coolant Injection Available through 10 hours via HPCI/RCIC; reactor may be depressurized to 150 psi.

(This sensitivity performed assuming the installed diesel generators are unavailable).

FIGURE 3.1b

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)

JRHJ



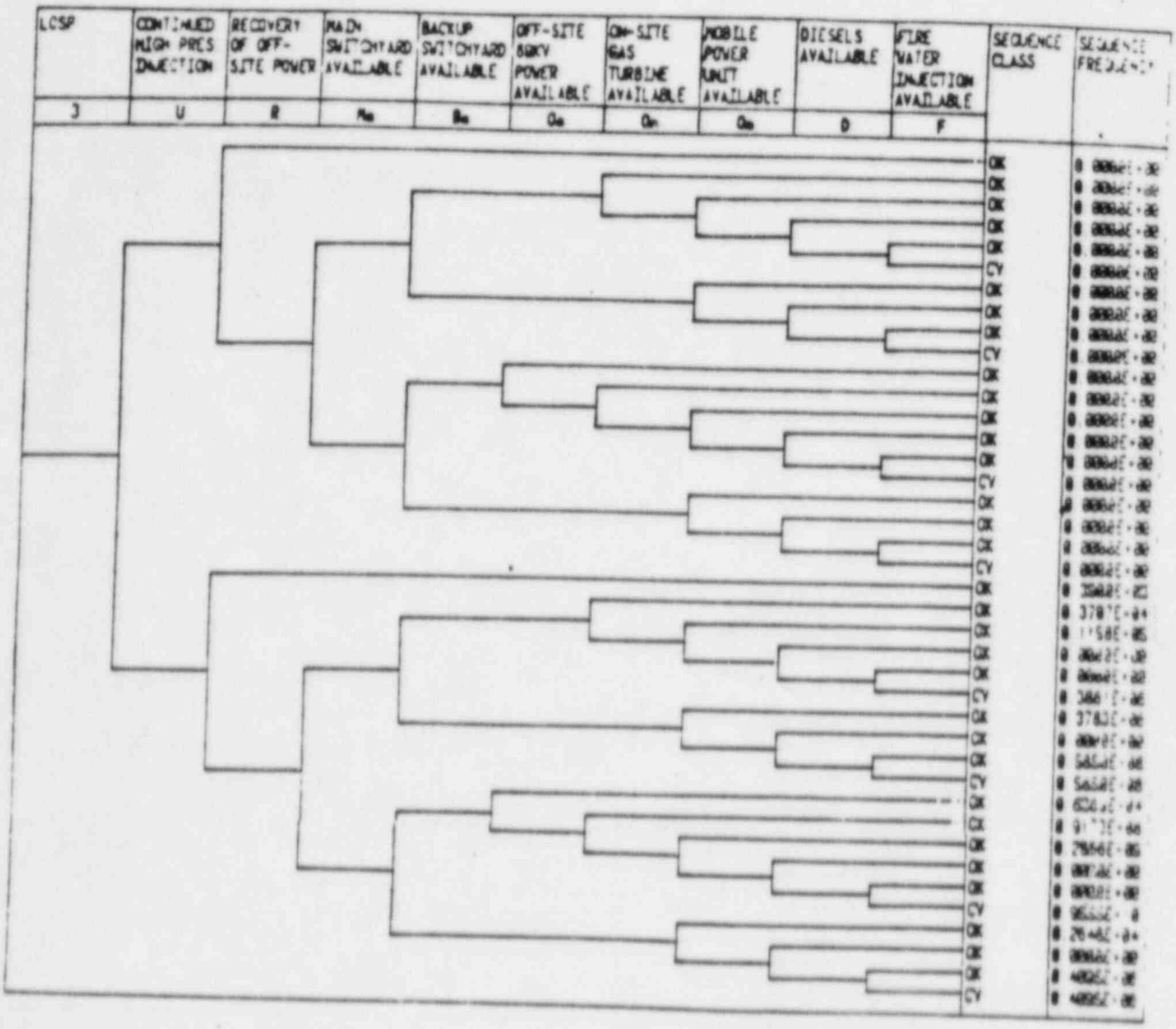
*Entry Conditions Sequence Type 2 : LOSP; Isolation; Reactor Scrammed; Reactor Integrity Intact; Coolant Makeup Available 0-4 Hours; Reactor may be depressurized to 150 psia.

(This sensitivity performed assuming the installed diesel generators are unavailable).

FIGURE 3.1c

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)

JRH1

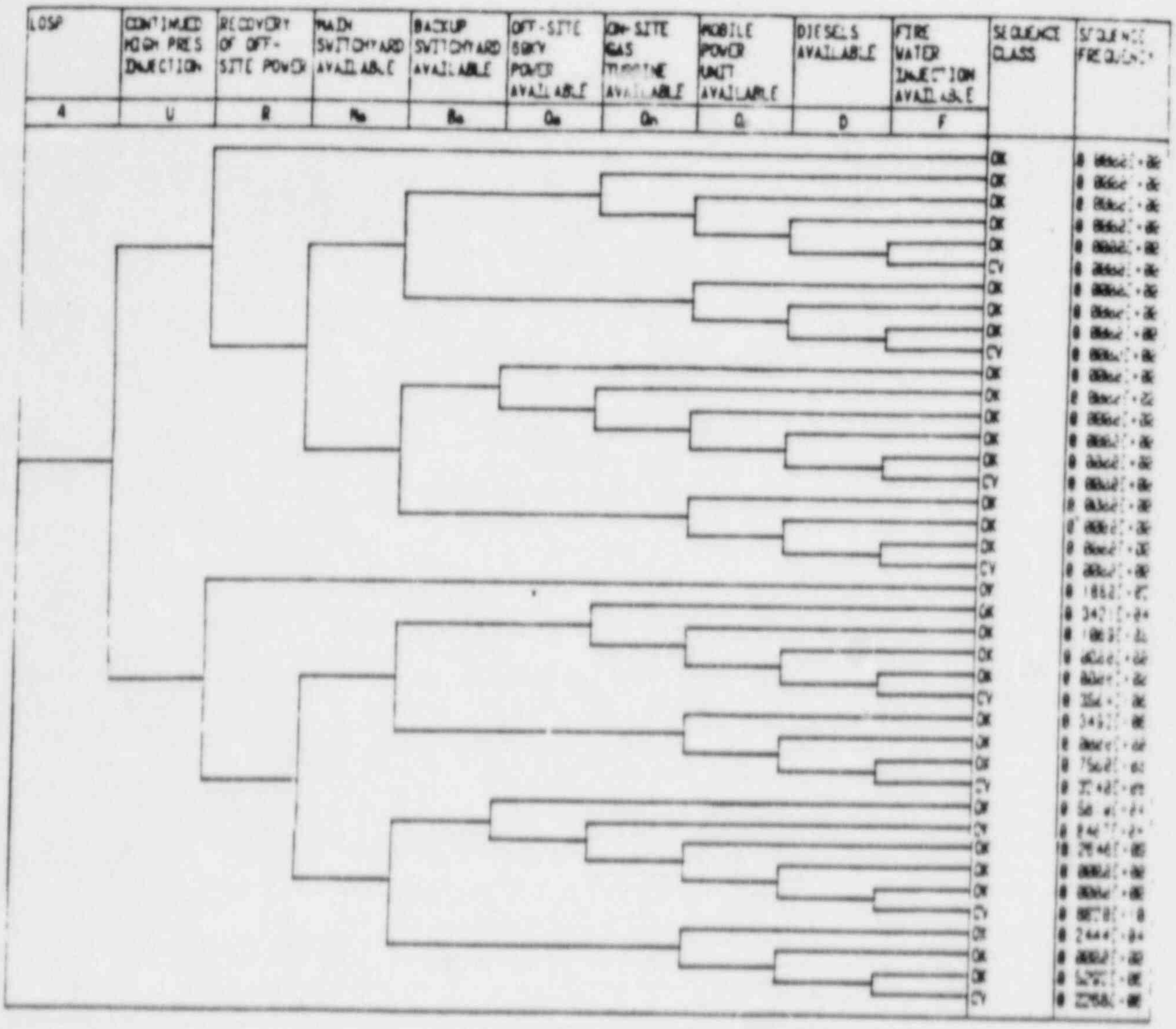


*Entry Conditions Sequence Type 3 : LOSP; Isolation; Reactor Scrammed; SORV, LOCA or ADS; no Coolant Makeup Available; Reactor Depressurized to Less than 65 psia.

(This sensitivity performed assuming the installed diesel generators are unavailable).

FIGURE 3.1d

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)

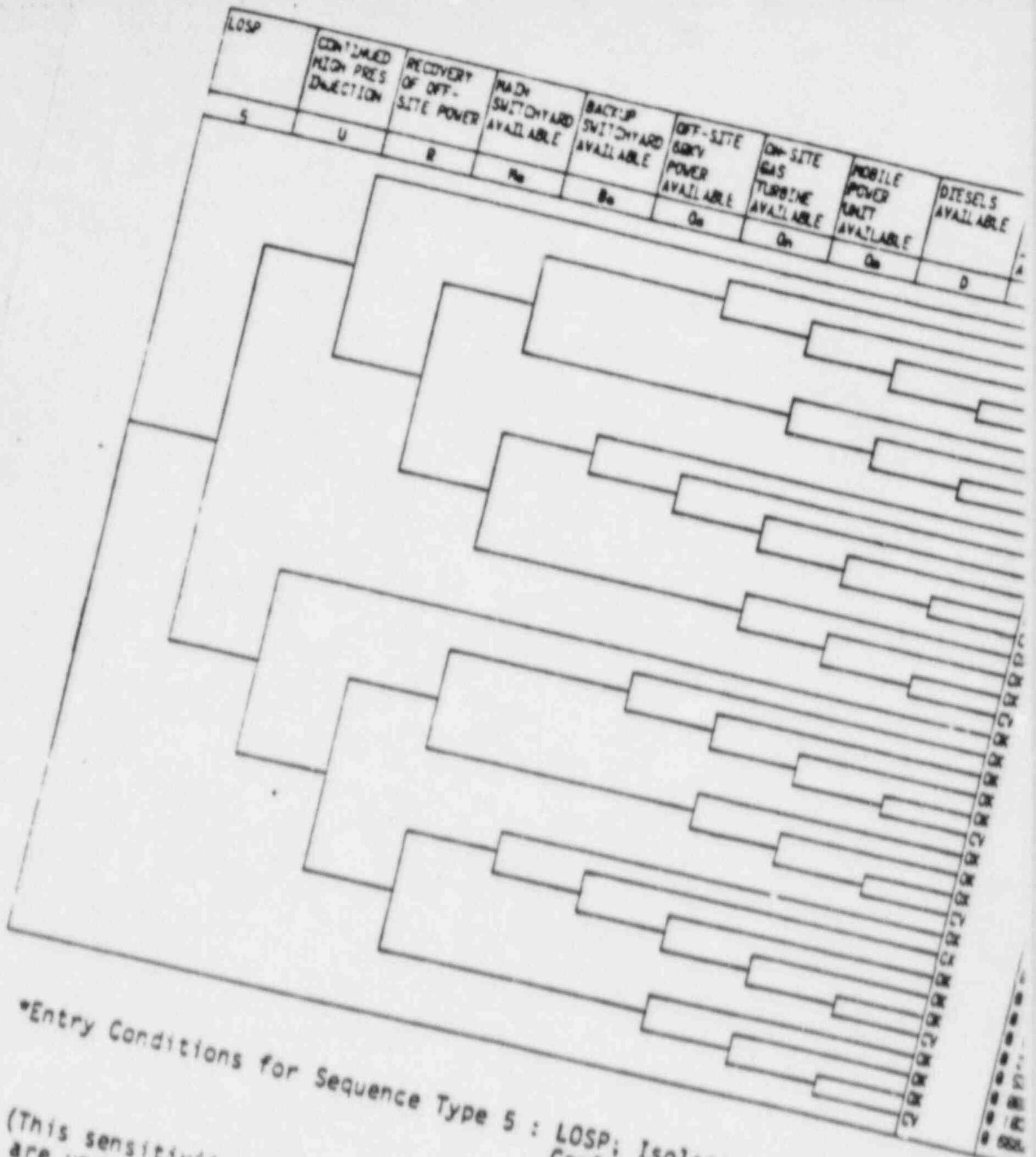


*Entry Conditions Sequence Type 4 : LOSP; Isolation; SORV; Coolant Injection Available Initially.

(This sensitivity performed assuming the installed diesel generators are unavailable).

FIGURE 3.1e

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)



*Entry Conditions for Sequence Type 5 : LOSP; Isolation; no initial Coolant Makeup; Procedural Depressurization.

(This sensitivity performed assuming the installed diesel generators are unavailable).

FIGURE 3.1f

LOSS OF OFFSITE POWER INITIATED EVENT TREE: 5% INITIAL POWER (EQUILIBRIUM)

JRH1

3 hours. The quantification of the conditional probability of an SORV or LOCA is based upon the following:

- SORV
  - A single challenge to the first bank of SRVs.
  - The failure probability of the SRVs from the Shoreham PRA [3-1] increased by 100% to account for the wear-in period of operation.
  - The operator action to prevent SRV cycling through running of the turbine driven systems (HPCI or RCIC) or opening a single SRV manually. The use of the turbine driven systems is judged the preferred course of action and is an integral part of the operator training. .
- LOCA: the conditional probability of a small LOCA induced during the loss of offsite power, including recirculation pump seal failures is included in the estimate of failure of primary system integrity.

The result of this evaluation is an estimate of the conditional probability of a failure of primary system integrity of .03/event.

High Pressure Coolant Injection (U): The Shoreham plant has the capability to provide coolant injection without AC power through the HPCI and RCIC turbine driven systems. If one of the turbine driven systems is available to provide the initial coolant injection following a scram, then the time to a core vulnerable condition is substantially increased. The calculations summarized in Appendix B indicate that the time available prior to core uncovery following one cycle of HPCI/RCIC operation can be on the order of 3 days depending upon the operator response regarding reactor depressurization (See below).

Adequate coolant inventory makeup using the turbine driven systems may be required under a variety of conditions if a station blackout were to occur. Figure 3.1a separates out the principal types of challenges as follows:

Type 1: The primary system integrity intact; coolant injection available through at least 10 hours.

Type 2: The primary system integrity intact; coolant makeup available initially.

Type 3: A breach of primary system integrity, no coolant makeup available.

Type 4: A breach of primary system integrity; coolant makeup initially available.

Type 5: No initial coolant makeup and procedural depressurization.

In a manner analogous to the Shoreham PRA full power analysis, coolant injection availability has been treated in a time dependent fashion. This time dependency arises from the potential for time dependent failure modes induced by such occurrences as battery drainage or high room temperatures. Therefore, Figure 3.1a provides two coolant injection phases U' (0-4 hours) and U (4-10 hours). The quantification of the success probability at each of these sites is similar to that of the full power PRA with the exception that the system availabilities have been decreased due to the wear-in effects discussed in Section 3.3. It should be pointed out that LILCO analysis has been performed to show that HPCI/RCIC operation can provide adequate coolant injection for more than 24 hours with proper DC load stripping from the batteries [3-4]. Therefore, this possibility has been included in the analysis in a probabilistic fashion.

Situations requiring the high pressure coolant injection function early in time involve cases with a SORV or LOCA. If HPCI* is initially available to provide coolant injection, but fails during subsequent demands, then the reactor coolant inventory will be adequate for at least 30 hours. If HPCI and RCIC are

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*RCIC is marginal for this task because depressurization occurs sufficiently rapidly to preclude adequate coolant injection, however more detailed calculations could show its capability is adequate also.

unavailable to provide this initial injection, then the time to potential core vulnerable conditions is calculated to be as low as 3 hours.

Another postulated condition is that associated with a failure to scram following LOSP. For such situations the MSIVs will close and the SRVs will open to remove the power being generated. There is no substantial effect of an SORV on HPCI/RCIC availability; rather, it is postulated to result in a stable reduced reactor pressure (i.e., 350-400 psia). Under such conditions, the high pressure coolant injection system (HPCI and RCIC) would be required to provide coolant makeup in a relatively short period of time. It is found that RCIC alone is probably acceptable for maintaining adequate coolant injection - however, it is not given credit in this analysis. Therefore, if HPCI is unavailable it is judged that core vulnerable conditions would occur within 3 hours. If HPCI is successful then coolant injection can be maintained for an extended period of time, which is limited by the interaction between the containment response and the HPCI system capability. The time to reach an elevated suppression pool temperature, i.e., approximately 240°F is more than 3-4 hours.

In addition, there is a feedback mechanism which must be accounted for: by procedure the operator is directed to depressurize the reactor (less than 100°F/hr). A result of this depressurization process may be to defeat the HPCI and RCIC systems. This is modeled in the event tree as function Z discussed below.

Reactor Depressurization (Z,X): The status of reactor pressure is important to the scenario development because:

1. Depressurization of the reactor will result in a loss of coolant inventory.

2. Depressurization may also occur to such an extent as to disable both HPCI and RCIC.

3. Accident sequence timing during a station blackout is strongly dependent on the rate at which RPV coolant is discharged.

Because of these important considerations, the event tree (Figure 3.1a) includes two functional events for depressurization:

Z: This event represents a rapid emergency depressurization equivalent to ADS actuation. While the operator is warned to avoid this, it has such an important influence on the course of the postulated sequence as to require separate consideration. The impact of rapid depressurization (ADS) is to uncover the core and defeat HPCI and RCIC. This has the effect of reducing the time* available to recover AC power. (It appears that it might even be prudent to disable ADS during start-up testing.)

X: This event reflects the required procedural step of controlled depressurization following a station blackout (less than 100°F/hr). The character of the controlled blowdown is such that HPCI and RCIC can be maintained as viable coolant injection sources. Therefore, the reason for considering this functional event is only to assess the cases where the HPCI/RCIC may fail. In such cases, the coolant inventory could be discharged during depressurization at a faster rate than decay heat boil-off. Under such circumstances (no HPCI/RCIC; slow depressurization) the time * to recover AC power before a core vulnerable condition would be decreased.

The remaining functional events appearing in Figures 3.1 b,c,d,e, and f are as follows:

Recovery of Offsite Power (R): Depending on the accident sequence there may be varying amounts of time available to the operator to restore offsite power. Specifically, the recovery times and conditional probabilities from Table 3.1-1 apply. The probabilities from Table 3.1-1 are consistent with those used in the Shoreham PRA and are also quoted in Appendix A.

*Reduction in time available translates directly into decreased probabilities of recovery of AC power.

Main Switchyard Available (Ms): As shown in Appendix D, main switchyard faults are a large contributor to the extended LOSP sequences. Based upon generic nuclear plant operating experience, a point estimate conditional failure probability of 0.7/event is used for the LOSP failure mode which involves the main switchyard.

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Backup Switchyard (Bs): As shown in Appendix D, there may be common mode failures which would adversely impact both the main and 69KV backup switchyard. For the initial start-up testing at Shoreham the conditional probability of the backup switchyard failure of 0.3/event is judged to be conservative because:

- The Shoreham backup switchyard is physically separated from the main switchyard.

- The backup switchyard will be required to be available during power operation.

- Power operation will not be performed during periods of severe weather, which have characteristically been the cause of the observed common cause switchyard related failures.

Offsite 69KV Power Available (Os): Because of the high level of redundancy of the offsite power grid and its restoration ability, it is found that for those LOSP initiators which are generated by failure modes at the plant, i.e., switchyards, that the offsite power sources will remain viable with a high reliability. Therefore, the ability to restore offsite power to the plant electrical busses is principally a function of the availability of an on-site switchyard, ie., main or 69KV backup.

On-Site Gas Turbine Available (On): LILCO has performed a detailed reliability evaluation of data from the on-site blackstart gas turbine. The results of the analysis indicate that the unit reliability is 0.96/event. The unit can be used in those accident sequences in which the backup switchyard remains available.

TABLE 3.1-1 CONDITIONAL PROBABILITY OF RECOVERY OF OFFSITE POWER AS A TIME DEPENDENT FUNCTION

ACCIDENT SEQUENCES	INITIATOR FREQUENCY (PER RX YEAR)	TIME AVAILABLE FOR LOSP RECOVERY (HRS)	CONDITIONAL PROBABILITY OF FAILURE TO RECOVEP OFFSITE POWER
TYPES			
1)	6.2E-2	60	1.E-4**
1)'	6.2E-2	48	1.E-3**
2)	1.5E-2	30	5.E-3
3)	5.2E-4	3	.25
4)	2.0E-3	10*	.06
5)	2.1E-3	7.5	0.13

* Based on containment conditions.

** Estimates of recovery probability at times greater than 24 hours are speculative since insufficient data exists to characterize such recovery probabilities even on a generic basis.

Does the .06 [illegible] take credit for deletion of severe weather caused LOSP events? I think no. Cf p 3-107 [illegible]. However, should really correct, [illegible] of PRA. However, should really correct, freq of ~~[illegible]~~ [illegible] where power is lost for, say [illegible]. Effect greater for longer outages.

Mobile Power Units Available (Om): As temporary measures to further improve on-site power source reliability, LILCO has installed four mobile diesel generator units. These units bypass both switchyards and can be manually switched into the plant when required. A detailed reliability analysis of these units by LILCO indicates a reliability of .93 per diesel generator. However, because of the need for operator interaction to start and connect the units, a conditional failure probability of .03 is used in the analysis to characterize the common cause failure of 3 out of 4 of the mobile units (any 2 diesel generator units are assumed required for minimum load). These may be very conservative assumptions. N.B.

Diesel Generators Available (D): For this analysis, the installed diesel generators are assumed as a groundrule to have an unavailability of 1.0.

Fire Water Available (F): LILCO has performed an analysis of additional backup reactor core cooling modes which do not require AC power. These methods require extensive manual operation and unusual plant lineups and therefore are heavily time dependent. These alignments have been practiced by the operating staff at Shoreham to verify their workability. In addition, since they are heavily operator dependent, there may be a common mode coupling between these actions and those associated with the manual switching of the mobile diesel generators and the black start gas turbine. The conditional failure probability of utilizing the fire water sources as a coolant injection back up are taken to be time dependent as follows:

Time Available	Conditional Failure Probability
2 days	0.1
30 hrs.	0.2
10 hrs.	0.3
7.5 hrs.	0.4
3 hrs.	0.5

JRH1

3.1.2 Timing

The LOSP event tree evaluation is based upon the accident sequence timing from a number of deterministic calculations, including:

a. General Electric LOCA calculations
b. LILCO calculations
c. Appendix B calculations with MARCH

Since operator action and restoration of power are strong time dependent functions, the calculated timing of events sets the boundary conditions for the accident sequence quantification. Table 3.1-2 summarizes some of the important timing for the LOSP initiated accident sequences.

3.1.3 Summary of Dominant Sequences

Based upon the calculated timing and the quantified accident sequences it is found that the following sequences dominate the core vulnerable frequency:

- o Station blackout scenarios in which high pressure coolant injection (HPCI/RCIC) is unavailable and the operator slowly depressurizes according to procedure, and no AC power can be restored within 7.5 hours.

- o Station blackout plus a SORV causes loss of both coolant inventory and HPCI injection. Why does HPCI fail?

3.1.4 LOSP - Sensitivity Study

3.1.4.1 Two SORVs

The case of two SORVs has not been explicitly calculated in the 5% power restricted case since the conditional probability of two SORVs is calculated to be quite low, i.e., approximately 1E-5 per transient challenge for LOSP initiators from 5% initial power. The consequences of the two SORV case would

be similar to a large LOCA and therefore the addition to the core vulnerable frequency for LOSP initiation would be approximately:

a) Initiator frequency (LOSP 2 hours) = 4E-2/RxYr

b) Two SORVs = 1E-5/event

c) Auxiliary on-site power sources = 1E-1/event

This yields a total addition to the core vulnerable frequency of less than 4E-8/reactor year.

3.1.4.2 Diesel Generator Reliability

The probabilistic quantification has been carried out assuming that the installed diesel generators are not included in the plant design. However, a sensitivity study performed on the diesel generator reliability has shown that even if the joint reliabilities are arbitrarily reduced by two orders of magnitude below that derived from generic data, the installed diesel generators still provide significant reduction in the calculated core vulnerable frequency for the loss of offsite power initiated accident sequences. The calculated reduction is on the order of a factor of 3 to 10 (see Appendix E).

3.1.5 Results of LOSP Initiator

The Shoreham PRA calculated the frequency of core vulnerable events resulting from loss of offsite power initiators and determined the total frequency to be approximately 1E-5 reactor year. This frequency is judged to be sufficiently low as to be acceptable both in relationship to other accident sequence frequencies and relative to the proposed NRC safety goal.

The calculated core vulnerable frequency due to accident sequences initiated by a LOSP for Shoreham during the start-up test phase with the installed diesel generators assumed unavailable is approximately 3E-6/reactor year (see Table 3.1.3). This frequency is less than the comparable sequence frequencies for Shoreham when operating at full power with installed diesel capacity evaluated

TABLE 3.1-2 CONSTRUCTION OF TIMING SCENARIOS FOR THE CORE VULNERABLE STATES (5% POWER RESTRICTION)

TIMING

PLANT CONDITIONS	ACCIDENT SEQUENCE DESIGNATOR	TIME DELAY PRIOR TO BOIL OFF INITIATION	BOIL OFF INVENTORY (TIME TO UNCOVER)	TIME TO CORE HEAT UP	TOTAL TIME AVAILABLE PRIOR TO CORE VULNERABLE
1) Station blackout, with 1)' initial coolant make-up through 10 or 30 hrs. as noted	1) Te, Tex	10 hrs.	>33 hrs.	15 hrs.	Approximately 2 days
	1)' Te, Tex	30 hrs.	≥35 hrs.	16 hrs.	Approximately 3 days
2) Station blackout, with makeup available through 4 hrs	2) TeU4-10 (Controlled Depressurization)	4 hrs.	30 hrs.	13 hrs.	45 hrs.
2)' Loss of makeup initially but no depressurization	2)' TeUX (High Reactor Pressure)	0	18 hrs.	12 hrs.	30 hrs.
3) Station blackout, ADS, no coolant injection	TeZ	0	30 min.	2½ hrs.	3 hrs.
4) Station blackout SORV, One HPCI injection to Level 8	TeP	0	25 hrs.	12 hrs.	>35 hrs
5) Station blackout depressurization, No make-up available	TeU'	30 min.	45-60 min.	6-7 hrs.	7.5 hrs.

with generic failure rates, and is far below the proposed safety goals. In addition, if even modest allowance is provided for the installed diesel generator this low frequency is further reduced by an additional factor of 3 to 10.

TABLE 3.1.3 SUMMARY OF LOSP
ACCIDENT SEQUENCE QUANTIFICATION

LOSP SEQUENCE TYPE	TIME TO CORE VULNERABLE	FREQUENCY (PER Rx YR) DIESELS PROB-ABILISTICALLY AVAILABLE	FREQUENCY (PER Rx YR) DIESELS UNAVAILABLE
TYPE 1	2 days	1.E-8	1E-7
TYPE 2	30 hours	3.2E-8	3.2E-7
TYPE 3	3 hours	8E-8	8E-7
TYPE 4	10 hours	5.9E-8	5.9E-7
TYPE 5	7.5 hours	1.5E-7	1.5E-6
TOTAL		3.3E-7	3.3E-6

3.2 LOSS OF COOLANT ACCIDENTS (LOCAs)

This section provides an assessment of plant risk associated with postulated LOCAs initiated from plant operation at 5% power. A variety of LOCAs are analyzed in the PRA because each represents a unique, potentially severe, challenge to several plant safety systems. In general, LOCAs have their largest impact on the availability of coolant injection systems, but there is also the potential for induced failures in containment systems. These challenges to containment, which are shown in the Shoreham PRA not to have a significant impact on the frequency of core vulnerable conditions, nevertheless may have a

significant impact on overall risk estimates because of the potential for direct radionuclide releases outside of containment. This aspect of the PRA is one of the considerations for the analysis of LOCAs initiated during operation at 5%. The discussion to follow includes estimates of LOCA sequence timing, a requantification of LOCA initiated core vulnerable sequences, and a summary of the overall risk associated with LOCAs at low power.

3.2.1 Sequence Timing for LOCAs Initiated From 5% Power

The five types of LOCAs analyzed in the PRA include large, medium, and small breaks inside of containment, RPV ruptures, and large LOCAs outside of containment. Because of the low decay heat energy levels following a scram from low power, the timing of postulated accident sequences associated with these initiators takes on a different character from that described in the PRA (See Appendix B). The reference scenario for the evaluation of sequence timing involves the double ended rupture of main steam line piping with subsequent failure of all injection. The initial blowdown phase as the RPV depressurizes to containment pressure is judged to be essentially equivalent to the blowdown described for LOCAs initiated from full power operation. However, for low power operation, the core heatup phase following a large LOCA is expected to be significantly longer, i.e., the onset of high fuel clad temperature can be delayed for up to three hours. This reference value of 3 hours is one of the key parameters used to requantify sequence frequencies for large LOCAs in the following section. In addition, this sequence timing is applied to medium LOCAs as well. Small LOCAs are assumed to be characterized by sequence timing similar to that described for transients in which the reactor is isolated (Section 3.3). While this may not be conservative, it does not appear to play a significant role in the calculations. Given that containment survives the initial LOCA blowdown and RPV makeup is established, the time available prior to containment overpressure is greatly extended (on the order of 20-30 days) because of the lower decay heat levels. As described in Section 3.3, sequences of this nature are judged to be small contributors to core and containment vulnerable conditions. Therefore, a detailed quantitative analysis of these sequences is not presented in this section.

3.2.2 Quantification of LOCA Sequences Initiated From 5% Power

In this section, accident sequences for each of the five LOCA initiators are reviewed and, where possible, requantified to reflect the frequency of core vulnerable conditions due to operation at low power. Each of the initiator frequency estimates is derived from very limited data. Therefore, factors which may contribute to the initiator frequency estimates, such as the low number of stress cycles placed on the reactor coolant system, are not quantifiable. For this reason, the LOCA initiator frequency estimates are judged to be the same as used in the Shoreham PRA.

3.2.2.1 Large LOCA and RPV Rupture Sequence Quantification

There are three significant events to consider in the large LOCA sequence quantification: reactor scram (Event C), vapor suppression system availability (Event D), and low pressure injection (Events V', V", and V'''). Failure of any one of these three events is assumed to lead to core vulnerable conditions in the PRA. The first two of these events involve a challenge similar to that expected for LOCAs initiated from full power operations. Therefore, no requantification of events C and D, is considered. Failures of low pressure injection, however, may be backed up by various operator initiated contingency actions. Injection of service water directly into the RPV is one such contingency. The use of the fire water system, which can be connected through a spoolpiece to the ultimate core cooling connection, is another such contingency which can be implemented within a 3 hour time frame. Because of the potential for significant delays in performing these complex contingency actions, a relatively high conditional failure probability is estimated for operator backup actions: 0.5/demand. Based on this reduction in conditional failure probability coupled with the assumed decrease in reliability due to a wear-in-period, the total failure probability of Event V remains similar to the original PRA. The accident sequence requantification is summarized in Table 3.2-1.

In addition to the large LOCA accident sequences another set of sequences involves breaches of the RPV. For break sizes beyond a DBA successful mitigation may still be possible from initial powers of 5% if the core spray

system is functioning. Therefore, sequences involving RPV breach and unavailability of adequate coolant injection are found to make a negligible contribution to the frequency of core vulnerable states due to loss of coolant makeup.

3.2.2.2 Medium LOCA Sequence Quantification

The medium LOCA initiated accident sequences quantified in the PRA involve a delayed RPV depressurization in which the use of HPCI for short term injection eventually requires initiation of low pressure injection systems. In this regard, short term HPCI operation is considered a viable method of reactor depressurization, but necessitates low pressure system operation. Because of the very low decay heat levels following scrams from low power, a LOCA in this size range is assumed capable of depressurizing the reactor without HPCI or ADS prior to core heatup. In other words, it is assumed that the RPV depressurizes below the shutoff head of low pressure systems prior to the onset of core overheat. This assumption makes failures of Event X, timely depressurization, no longer relevant. Therefore, virtually all medium LOCA events depressurize sufficiently rapidly, and to such a degree that they require mitigation by low pressure injection systems. Additional credit for backup low pressure injection system capability similar to the large LOCA case accounts for a reduction factor of 2.0 in the core vulnerable frequency for cases involving failures of low pressure injection. However, the potential increase in system unreliability during the plant start-up (wear-in period) offsets this improvement. The impact of these assumptions on frequency estimates for medium LOCA accident sequences appears in Table 3.2-2.

3.2.2.3 Small LOCAs

Small LOCAs, as stated in Section 3.2.1, are judged to be characterized by depressurization rates substantially less than SORV sequences discussed in Section 3.3. One of the implications of this is that the rate of coolant loss is much lower, extending the time available for recovery actions and reducing the RPV makeup flow requirements. In recognition of these sequence characteristics, the success criteria for RPV inventory makeup is modified to incorporate the viability of the CRD pumps as an adequate coolant injection

source. In addition, the fire water and service water systems contingency actions previously described are also considered successful options. In the first case, a positive point is that CRD pump flow is available at any reactor pressure. A second consideration, however, is that the break location for a small LOCA may be in the lower region of the RPV which may divert a significant fraction of CRD flow. Therefore, an estimated reliability of 0.9/challenge is applied to small LOCA sequences to account for CRD flow. Third, RPV depressurization is judged inevitable in the long term due to the low decay heat levels. Therefore, HPCI, RCIC, and feedwater are not given credit for long term operation. This impacts a significant fraction of sequences which were previously considered successful recovery states; increasing the challenge to low pressure systems. Additional reliability of 0.5/challenge is used to account for operator backup contingencies in sequences involving failures of low pressure injection systems, which offsets the factor of 2 increase in equipment unreliability due to the start-up (wear-in phase). Table 3.2-3 summarizes the accident sequence quantification for the high and low power cases.

3.2.2.4 Large LOCAs Outside of Containment

Large LOCAs outside of containment from 100% power are extremely rare events, which by their nature may cause failures of RPV coolant injection systems. The situation for cases of low initial power may be substantially different. There are anticipated to be more success paths due to additional time available; however, no requantification has been performed because of the very low core vulnerable frequency.

3.2.3 Impact of LOCA Requantification on PRA

Table 3.2-4 summarizes the quantification results for the LOCA sequences providing a comparison between operation at high and low power. As shown, the total estimated frequency of core vulnerable sequences due to LOCAs decreases by approximately a factor of 4.

TABLE 3.2-1

LARGE LOCA SEQUENCE FREQUENCY ESTIMATES FOR

HIGH AND LOW POWER OPERATION

SEQUENCE	ACCIDENT CLASS	ESTIMATED FREQUENCY-FULL POWER (EVENTS/REACTOR YR)	ESTIMATED FREQUENCY LOW POWER (EVENTS/REACTOR YR)
AV	III	9.8E-8	9.8E-8
AD	III	7.0E-8	7.0E-8
AC	III	7.0E-9	1.4E-8
AW	II	3.4E-7	*
AV'V"W	II	3.5E-7	*
$R_0R_1R_2$	III	3.3E-7	*
TOTAL, CLASS	II	6.9E-7	*
TOTAL, CLASS	III	1.8E-7	1.8E-7
TOTAL		8.7E-7	1.8E-7

*Estimated frequency less than 10^{-9}/reactor year

TABLE 3.2-2

MEDIUM LOCA SEQUENCE FREQUENCY ESTIMATES FOR HIGH AND LOW POWER OPERATION

SEQUENCE	ACCIDENT CLASS	ESTIMATED FREQUENCY-FULL POWER (EVENTS/REACTOR YR)	ESTIMATED FREQUENCY-LOW POWER (EVENTS/REACTOP YR)
S_1QUX	III	2.4E-7	N/A
S_1QUV	III	2.5E-7	2.5E-7
S_1GOL	III	5.0E-9	5.0E-9+
S_1OUW	II	7.5E-8	*
S_1QUV'W	II	2.1E-10	*
S_1QUV'V"W	II	1.9E-7	*
S_1C	IV	3.0E-8	3.0E-8+,++
TOTAL, CLASS	II	2.7E-7	*
TOTAL, CLASS	III	5.1E-7	2.6E-7
TOTAL, CLASS	IV	3.0E-8	3.0E-8
TOTAL		8.0E-7	2.9E-7

* Estimated frequency less than 10^{-9}/Reactor Year

\+ Additional time to core vulnerable could result in a reduction in the calculated frequency.

++ The RHR heat removal capability of approximately 3% of full power could result in a significant change in the perception that a LOCA coupled with a failure to scram could not be effectively mitigated from low power.

TABLE 3.2-3

SMALL LOCA SEQUENCE FREQUENCY ESTIMATES FOR HIGH AND LOW POWER OPERATION

SEQUENCE	ACCIDENT CLASS	ESTIMATED FREQUENCY-FULL POWER (EVENTS/REACTOR YR)	ESTIMATED FREQUENCY-LOW POWER (EVENTS/REACTOR YR)
S_2QUX	III	1.5E-8	2.9E-9
S_2QUV	I	1.1E-10	5.0E-9
S_2GOL	I	2.1E-7	2.1E-8
S_2W	II	1.3E-8	*
S_2QW	II	1.1E-10	*
S_2QUW	II	2.6E-7	*
S_2QUVW	II	1.1E-8	*
TOTAL, CLASS	I	2.1E-7	2.6E-8
TOTAL, CLASS	II	2.1E-8	*
TOTAL, CLASS	III	1.5E-8	2.9E-9
TOTAL		2.6E-7	2.9E-8

*Estimated frequency less than 10^{-9}/ Reactor Year

TABLE 3.2-4

FREQUENCY ESTIMATES FOR LOCAs IN PRA ACCIDENT CLASSES

ACCIDENT CLASS	ESTIMATED FREQUENCY 100% POWER (EVENTS/REACTOR YR)	ESTIMATED FREQUENCY 5% POWER (EVENTS/REACTOR YR)
I	2.1E-7	2.6E-8
II	9.9E-7	7.2E-9
III	1.0E-6	4.4E-7
IV	3.7E-8	3.0E-8
V	3.6E-8	3.6E-8
TOTAL	2.3E-6	5.4E-7

3.3 OTHER TRANSIENTS

Loss of offsite power initiated transients are evaluated in Section 3.1. The category of "other transients" includes all of the remaining events described in the Shoreham PRA in which the reactor is challenged and successfully scrammed. Thus, this category includes both relatively high frequency "anticipated" transients such as spurious trips, MSIV closure, etc. and low frequency initiating events as shown in Table 3.3-1.

TABLE 3.3-1
SEQUENCE INITIATORS IN WHICH THE REACTOR IS SCRAMMED AND THE PRIMARY SYSTEM IS INTACT

ANTICIPATED TRANSIENTS+	LOW FREQUENCY INITIATING EVENTS
Turbine Trip	125V DC Bus Failure
Manual Shutdown	Reactor Building Internal Floods
MSIV Closure	Reactor Water Level Instrument Failures
IORV	Loss of Reactor Building Service Water
Loss of Feedwater	
Loss of Condenser Vacuum	

+ LOSP treated in Section 3.1

This section of the report consists of a reassessment of the dominant core vulnerable sequences (as defined in the PRA) which are associated with these transients for start-up operation when power is restricted to a maximum of 5% of full power. By definition, these initiators are contributors to the frequency of accident Classes I and II which involve loss of adequate coolant inventory and unavailability of containment heat removal, respectively (see Section 2).

Before beginning the detailed quantification, it is useful to provide some perspective on the effects of low power operation on the postulated accident sequence Classes I and II. The evaluation of core or containment vulnerable conditions associated with the unavailability of adequate containment heat removal (Class II) includes the following considerations:

- The capacity of the suppression pool is very large (approximately 600,000 gallons). For those events resulting in the transfer of primary system stored energy into the suppression pool, the initial increase in suppression pool temperature would be on the order of 50°F, raising the temperature to approximately 150°F. The decay heat load for the next day would add about another 10°F. Additional time without containment heat removal would result in progressively slower temperature rises. This yields an extensive time available to the operating staff to respond with extraordinary means if deemed necessary. Since the Shoreham LPCI pumps can effectively pump suppression pool water even when saturated, these sequences are found to be well below frequencies which can be credibly quantified.

- The time required for initiation of containment heat removal following a shutdown from 5% power is conservatively calculated to be on the order of 10 to 30 days.

- In addition, there is significant evidence that even at these times the passive heat losses from the containment through the containment walls will exceed the decay heat produced in the fuel.

Given the extended amount of time to reach a positive equilibrium, it is found that the containment and core vulnerable frequency derived from these sequences (i.e., TW) are below 10^{-9}/reactor year. Based upon this assessment, the focus of the evaluation of the transient initiators is on the ability to maintain adequate coolant inventory in the reactor vessel. (The evaluation of transients coupled with a failure to scram is addressed in Section 3.4).

APPENDIX A

DOCUMENTATION OF INPUT DATA FOR PROBABILISTIC EVALUATION

A.1 SUMMARY OF LOSS OF OFFSITE POWER RECOVERY MODEL

The conditional failure probability to restore offsite power given that it has become unavailable is an important parameter in the evaluation of loss of offsite power accident sequences. The change in the probability as a function of time can be characteri:ed with reasonable accuracy over the initial 2 to 4 hours time period. Past this time the recoverability function becomes uncreasingly difficult to quantify due to the lack of adequate data. Virtually all of the data points used to characterize extremely long duration loss of offsite power situations are suspect. Based on this the EPRI data can be characterized as conservative. In order to establish a point estimate for recovery beyond 24 hours (i.e. the only data point), use is made of a model developed in the Zion PRA for offsite power recovery.

The Zion PRA summarizes the recovery of offsite power as follows:

"Very few events are expected to result in physical damage to all six of the transmission lines at Zion, such that none can be reenergized through remote switching operations". Some events (such as tornadoes, major ice storms, severe lightning, etc.) could, however, produce extensive damage, requiring several hours to repair. We express the following histogram as a representative distribution for the time to restore power to at least one 345 KV transmission line, given the fact that the cause of the fault has been identified and all required manual operations have been performed at the Zion switchyard. Available power line data (Reference 1.3-5) is consistent with this histogram.

Time Following Local Operations	Probability
0 - 1 minute	.70
1 - 10 minutes	.20
10 - 30 minutes	.05
30 - 60 minutes	.035
1 - 2 hours	.01
2 - 4 hours	.0045
4 - 8 hours	.0004
8 hours	.0001

The implementation of this model at Shoreham is done by coupling the data from EPRI NP 2301 with the Zion Model.

While this model does present a discontinuous function it is judged that the integrated effect of the model is reasonable. Specifically, the failures to recover probability at 24 hours is judged to be overly conservative and the recovery probability beyond 30 hours is unknown due to lack of data. Since extraordinary measures could be taken in the time frame greater than 24 hours, the integrated model assessment is assumed reasonable. The integrated model is shown in Table A.1.

Table A.1

SHOREHAM PRA VALUES FOR LOSS OF OFF-SITE POWER

				DATA SOURCE
LOSS OF OFF-SITE POWER INITIATING FREQUENCY (PER RX YR)		0.08 PER Rx Yr		LILCO GRID
CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFF-SITE POWER				
TIME PHASE	DURATION OF TIME PHASE (HRS.)	CUMULATIVE FAILURE PROBABILITY	CONDITIONAL FAILURE PROBABILITY+++	
I	0 - 2	0.52*	0.52	EPRI-2301 3/82
II	2 - 4	0.28**	0.54	
III	4 - 10	0.23+	0.82	
IV	10 - 24	0.06++	0.26	
V	30 hrs.	0.005++++		Engineering Judgement (Data Extrapolation)
VI	48 hrs.	0.001	0.2	
VII	96 hrs.	0.0001	0.1	

* Based upon recovery within 30 minutes.

** Based upon recovery within 2 hours.

\+ Based upon recovery within 4 hours.

++ Based upon estimated recovery within 10 hours.

+++ Conditional failure probabilities are constructed such that the condition failure probability of restoring off-site power in time phase N is contingent upon the probability of power not being restored in time phase N-1.

++++Conservatively derived from the zion PRA model which calculate values 10^{-4} to 10-3 for such events.

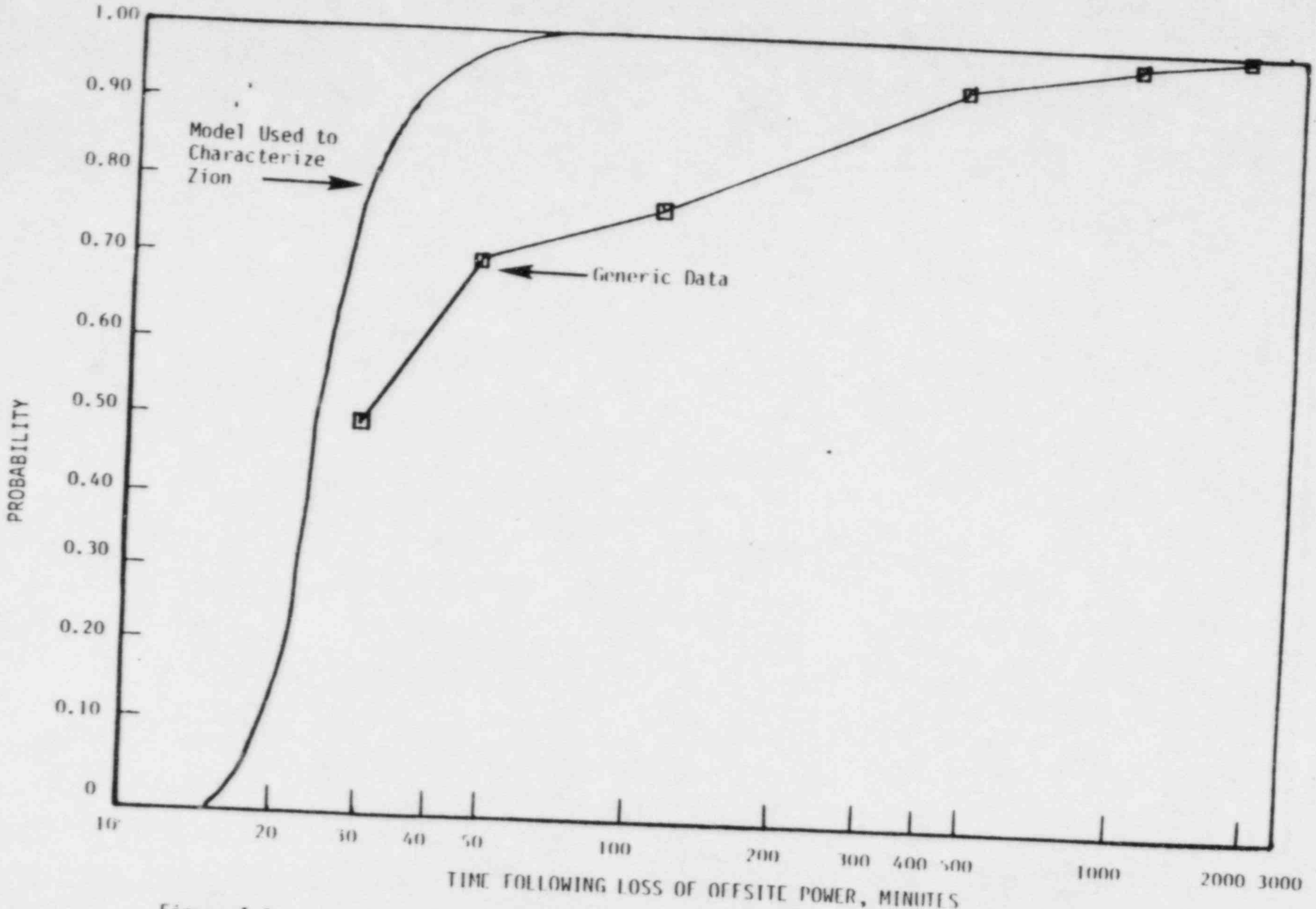


Figure A.1

Loss of Offsite Power Multiple Diesel Generator Failures
Total Time to Restore Offsite Power
(Cumulative Probability Distribution)

JRH1

APPENDIX B

PLANT RESPONSE DETERMINISTIC CALCULATIONS

PLANT RESPONSE DETERMINISTIC CALCULATIONS

B.1 Background

In the Shoreham PRA, [1], the risk methodology identified the important accident sequences which can lead to a core vulnerable condition and/or loss of containment integrity. The potentially risk dominant sequences were evaluated on the basis of an overall assessment of the risk to the public from these accidents initiated from full power and mature plant operation. The risk analysis considered both the estimated frequency and potential consequences of severely degraded core accident events. The risk dominant sequences evaluated in the Shoreham PRA, as in other PRAs, were found to be those leading to core meltdown and containment failure.

The quantification of the point estimate frequencies of these accidents centered on the initiating event frequencies and the assessed conditional failure probability of the required mitigating systems, given the initiating event. The results of the accident sequence evaluation in the PRA led to a quantification of five generalized accident sequence classes which represented a spectrum of degraded core accident events. These accident sequence classes were further assessed to lead to a range of off-site radionuclide release events. The radionuclide releases were found to yield potentially severe off-site consequences for those sequences where loss of containment integrity and core meltdown occurred early.

Because of the relatively short time scales of the characteristic plant response of the accident sequences initiated at full power operation, the evaluation of the consequences of potentially severe reactor accidents were bounded by the five generalized accident classes, for which the short term plant response (i.e., prior to core vulnerability) of specific accident sequences were not examined in detail. During low power operation, the short

term plant response of specific accident sequences become an important consideration in the overall probabilistic quantification process. Therefore, deterministic calculations of the plant response were performed for specific accident sequences which may be mitigated or recovered if initiated at low power. This appendix documents the deterministic calculations performed to investigate the plant response of several categories of accident sequences initiated at low power.

B.2 Low Power Operation Risk Analysis

The probabilistic risk analysis conducted here for the low power operation at Shoreham is basically a comparative evaluation. That is, the risk at low power is ultimately compared with the risk measure of full power operation as determined in the Shoreham PRA [1]. As such, it is important that the analysis of plant response, i.e., plant models and physical process models, are directly comparable between the PRA and this deterministic evaluation.

In evaluating the risk dominant sequences identified in the Shoreham PRA, as they apply to low power operation restricted to five percent power or below, the important consideration relates to: (1) the definition of the required plant systems which could successfully mitigate the accident, and (2) the time available for operator action within which recovery of failed systems or alternative systems may be used before core vulnerability occurs. These factors impact the assessment of the point estimate frequencies for those events leading to a core vulnerable condition.

In as much as the time constants of the accident progression during low power operation are critical in the probabilistic quantification of recovery and mitigative measures, a more recent physical process computational model, MARCH-2 [2] was used in these calculations. The Shoreham PRA used an earlier version of the same code, MARCH 1.1 [3], in the analysis of severe accident progression for radionuclide releases. It should be noted, however, that the use of MARCH-2 in this study does not conflict with the ground rules

established in the Shoreham PRA, since the present analysis is focused on the evaluation of the accident sequence timing, prior to core vulnerability for which the physical process models in the MARCH-2 code are judged to be more adequate and realistic.

Assumptions and Physical Process Models

In performing the analysis, the same plant models used in the Shoreham PRA for full power (as discussed in Appendix C of Reference [1]) were adopted in this study. The accident progression determinations are similar hence the assumptions and plant models are analogous. However, there are major differences which contribute to a substantial reduction in the risk associated with low power operation compared to the full power operation. In order to adequately address these differences, the following basic assumptions were made in this evaluation:

- o There is a linear relationship between radioactivity or decay heat and power, whereby the fission product inventory at 100% power is nearly 20 times that at 5% power [4].

- o The decay energy release rate following shutdown follows the standard curve for uranium fueled thermal reactors, assuming an equilibrium core. In addition, the spatial distribution at shutdown corresponds to that of the spatial distribution in the operating conditions at low power.

- o The effect of neutron capture in fission products and decay heat power from U 239 and Np 239 are included in the decay heat curve [5].

- o The decay heat power is related to the operating power history of the reactor core. The short operation time (30 days) and anticipated fluctuations of the power level during this time are not considered in this analysis. It is assumed that the reactor core is exposed to a constant power level of five percent for a period of two years for consistent treatment with the Shoreham PRA.

It is expected that the assumptions with regard to the decay heat rate will conservatively overestimate the reactor power after shutdown from low power operation. The impact of these assumptions will be examined in the sensitivity analysis of low power operation.

Because of the substantial reduction in the decay heat energy release rate, there is a reduction in the required coolant recovery rate for mitigating accidents at low power. Therefore the success criterion for the various accident types at full power analyzed in the Shoreham PRA could change. This analysis also investigates the possible success paths for other types of accident events which are potentially risk dominant.

Boundary Conditions

The evaluation of accident sequence timing requires information regarding core design, plant sytems, containment or other engineered safeguard systems, etc. and the definition of the initial reactor conditions. In this analysis, the reactor initial conditions are as defined by LILCO [6] for low power operation. Those conditions at low power operation which are significantly different from that at full power mature plant operation described in Appendix C of the PRA [1] were considered in the plant models. In particular, the following changes were made in the Shoreham plant models for this evaluation:

- The reactor core power profile reflects the low power operation fuel and control rod configuration with corresponding power peaking factors. The axial power peaking factors are summarized in Table 1 and the radial distribution is shown in Figure 1.

- Due to the lower power level, the average core temperature is initially lower at five percent than at the full 100 percent rated power. The initial core temperature used is 640°F for this analysis, which is approximately 400°F lower than the full power condition.

Table 1 AXIAL RELATIVE POWER PEAKING FACTOR PROFILE

Axial Node*	Peaking Factor
1	0.0388
2	0.162
3	0.239
4	0.305
5	0.3517
6	0.4208
7	0.5143
8	0.6376
9	0.7975
10	1.004
11	1.2802
12	1.4862
13	1.6252
14	1.7011
15	1.7233
16	1.7030
17	1.6517
18	1.5796
19	1.4959
20	1.4082
21	1.3228
22	1.2468
23	1.1667
24	0.8816
25	0.2597

* The axial node is identified incrementally relative from the bottom of the active fuel. Each node is 6 inches long for a total of 25 axial nodes representing the 12.5 foot high core active fuel region.

RELATIVE POWER DISTRIBUTION BY ASSEMBLY, ARRAY P(I,J)

	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.686	0.951	0.969	0.721	0.735	1.024	1.068	0.838	0.967	1.347	1.359	1.144	0.333
2	0.951	0.760	0.828	1.014	1.086	0.779	0.873	1.173	1.316	1.039	0.982	1.046	0.315
3	0.969	0.828	0.799	1.105	1.062	0.868	0.857	1.289	1.357	1.070	0.997	1.043	0.310
4	0.721	1.014	1.105	0.789	0.871	1.157	1.279	0.939	1.065	1.474	1.424	1.114	0.306
5	0.735	1.086	1.062	0.871	0.859	1.289	1.274	1.067	1.148	1.606	1.471	1.008	0.256
6	1.024	0.779	0.868	1.157	1.289	0.984	1.103	1.500	1.622	1.622	1.253	0.407	
7	1.068	0.873	0.857	1.279	1.274	1.103	1.131	1.469	1.507	1.367	0.502		
8	0.838	1.173	1.289	0.939	1.067	1.500	1.469	1.013	0.901	0.988	0.315		
9	0.967	1.316	1.357	1.065	1.148	1.622	1.507	0.901	0.688	0.673	0.195		
10	1.347	1.039	1.070	1.474	1.606	1.622	1.367	0.988	0.673	0.236			
11	1.359	0.982	0.997	1.424	1.471	1.253	0.502	0.315	0.195				
12	1.144	1.046	1.043	1.114	1.008	0.407							
13	0.333	0.315	0.310	0.306	0.256								

FIGURE 1 RADIAL RELATIVE POWER DISTRIBUTION AT LOW POWER OPERATION (1/4 CORE)

At low power operation, the reactor coolant system (RCS) pressure and temperature would be slightly lower. The reactor pressure of 950 psia versus 1020 psia is not expected to present a significant change in the RCS behavior particularly for the assessed dominant sequences involving reactor depressurization. Therefore, the basic PRA plant models were conservatively adopted with regard to the reactor pressure vessel (RPV) pressure and coolant temperature. The most important parameter change is related to the reactor core thermal energy. The stored energy of the core at low power is significantly lower than that at full power due to the lower initial average core wide temperature. This major difference is expected to affect the sequence timing determinations significantly, particularly where core dryout occurs shortly following the initiating event.

B.3 Assessment of Dominant Accident Sequences Initiated at Low Power

The analysis documented in this appendix focuses on those accident sequences which have been identified in the low power PRA as potentially dominant and for which operator action would be critical in mitigating such accident sequences. Three generalized categories of accident events with potentially similar impact on reactor response were evaluated. These include the following:

1) Transient events with the RCS isolated. Two cases of these isolation events were considered: a) no coolant injection and b) coolant injection occurs via the high pressure injection systems.

2) Transient induced loss of coolant accidents. The cases considered include a large LOCA, a stuck open relief valve and a controlled blowdown. The effect of coolant makeup was also examined for the case with a stuck open relief valve.

3) Transient events with failure to bring the reactor subcritical. The accident analysis primarily considered ATWS sequences where the coolant injection fails in the long term. This could be due to pump

failure from either of two causes: the containment integrity being lost resulting in adverse or harsh environment, or the lube oil degradation due to high suppression pool temperatures. In addition, the reactor response for an ATWS coupled with a SORV was also examined.

The categories of accident events described above were evaluated for the case where loss of off-site power occurs, hence the plant systems requiring off-site power were not included in the accident analysis. The evaluation consisted of MARCH-2 calculations supplemented by hand calculations of sequence timing perturbations.

B.3.1 Category 1 - Transient Events with the RCS Isolated

This category of accident events generally follow the Class I type of accident sequences addressed in the Shoreham PRA in which the reactor remains at high pressure. In this evaluation it is assumed that following a transient initiating event, the reactor is shutdown and isolated from the main condenser. The coolant inventory is boiled off and discharged to the suppression pool through the safety relief valves. Without sufficient coolant makeup, the core uncovers, the fuel would heat up and finally core vulnerability occurs. Following full power operation, the accident progression to core vulnerability is predicted to occur within 30 minutes.

For this accident sequence initiated at low power, there is a considerable time available between the initiating event and core vulnerability because of the low decay heat power level following shutdown at low power operation. The

accident analysis performed predicts several hours (approximately 18 hours) to initial core uncovering, and several more hours (approximately 11-12 hours) to heat up the core to core vulnerability*.

An assessment of this accident event category with coolant injection was made to determine the demands for the HPCI and RCIC operation and their potential impact on battery depletion for LOSP cases. The HPCI/RCIC pumps are steam turbine driven and start coolant injection when the RPV water level drops to the Level 2 set point. Coolant boil off to Level 2 was predicted to occur after approximately 12-13 hours. Following coolant injection and recovery of the water level to Level 8, the HPCI would terminate injection flow. It is assumed that these pumps would not be required to provide coolant makeup again until the coolant boils off to Level 2 (i.e., no operator intervention to throttle the flow is considered). Based on a calculation of mass and energy balance, heat up of the coolant inventory to the SRV set point and boil off to Level 2 was estimated to occur after approximately 1-1/2 days for this scenario, following the initial recovery of coolant inventory via the HPCI/RCIC pumps flow. Core uncovery would subsequently occur after an additional day. Therefore, the total estimated time to core uncovery given that the coolant inventory is recovered to Level 8 by a single HPCI/RCIC cycle is approximately 2-1/2 days.

It is apparent from these estimates that the time sequence of an isolation accident event can be dramatically changed by delaying initiation of RCS

* In these analyses, core vulnerability is defined in the context of cladding and fuel rod heat up (assumed as a lumped parameter) to a maximum nodal temperature of 1800°F to 2000°F where clad rupture could potentially occur. Incipient core melting could occur at 4130°F. In the Shoreham PRA, because of the relatively short time constants for core uncovery and overheat, core vulnerability was assumed to occur shortly after the core is uncovered to 2/3 of core height. Core recovery with subsequent core cooling is by no means precluded beyond this point.

coolant boil-off. In the accident sequence described above, it is assumed that there is no operator intervention to provide coolant injection prior to the automatic initiation of the HPCI/RCIC systems at Level 2. If coolant makeup is manually provided initially, but is ultimately lost, the time to uncover the core and core heat up would be delayed accordingly. Three such cases were evaluated. In these isolation sequence variations, it is assumed that coolant is provided with the water level in the reactor seeking Normal Water Level (NWL) to Level 8 and that the reactor pressure is maintained at 1130 psia, the lowest SRV setpoint. The time delays prior to boil-off initiation were assumed to be 4, 10, and 30 hours. The results of this evaluation are summarized in Table 2 below.

Table 2 - Summary of Impact on Isolation Sequence Timing for Various Time Delays

Isolation Sequence Variation	Time Delay Prior to Boil Off	Time To Core Uncovery	Time To Core Heatup
1	4 hours	> 30 hours	> 45 hours
2	10 hours	> 40 hours	> 55 hours
3	30 hours	> 60 hours	> 75 hours

An examination of the results shown in Table 2 indicates that the time to core heat up is not as significantly affected as the time to uncover the core for similar time delays prior to initiation of boil off. In isolation sequences, where the coolant is not lost rapidly, the core is still partially covered when the uncovered core regions begin to heat up. The uncovered fuel rods will experience cladding oxidation which could contribute to a significant portion of the energy required to overheat the core. The steaming rate is predicted not to be sufficient enough to cool the uncovered portions of the

core once elevated temperatures are reached which accelerate steam-cladding reaction. The radiative heat transfer models in MARCH-2, however, indicate that the radiative heat transfer losses may compete with the clad-water reaction energy release rate. For the sequence variation 3, involving a time delay of greater than 60 hours before core overheat occurs, the use of the radiative heat transfer models could extend the heat up time period of the fuel rods with the highest peaking factors by as much as 3-4 hours. In this analysis, a range of times is indicated to account for the uncertainty associated with the heat transfer models in MARCH-2.

These analyses did not consider the passive heat losses through the reactor vessel wall* to the drywell and the containment walls. It is judged that after two to three days, the decay heat rate would be very low compared to the passive heat losses through the vessel. This would tend to stop boil off and significantly reduce coolant loss from steam discharge through the safety relief valves. Therefore, it is possible that vessel inventory would be maintained with substantial reduction in the required injection capacity compared to full power operation. Such minimal injection requirements can be afforded by the CRD pump flow. Therefore, it was determined that an alternative long term injection success path for this category of accident events would be via CRD pump flow.

B.3.2 Category 2 - Transient Induced Loss of Coolant Accident

This accident event category encompasses the class of accident sequences involving loss of coolant inventory from the vessel due to RCS breach.

* An assessment of the heat transfer between fuel rods and from the fuel assemblies to the core shroud using the radiative heat transfer models in the MARCH-2 code indicates that core overheat after dryout would be extended significantly. Core vulnerability may not occur for isolation sequences where initiation of coolant boil-off is delayed substantially.

The methodology applied in this evaluation includes a requantification of each of the functions that appear in the transient event trees. However, it is not the intent of this analysis to reproduce the rationale used in the PRA, rather, it is intended only to emphasize the differences between high and low power operation which significantly influence quantitative estimates. Therefore, the approach involves applying scaling factors to initiator frequency estimates and event failure probabilities which are then applied to the dominant accident sequences to estimate the impact on the core vulnerable frequency.

3.3.1 Impact of Low Power Operation on Transient Initiated Accident Sequences

At the outset of this discussion, it is noteworthy to outline some of the sequence differences between high and low power operation. Specifically, the timing of the events is a key perspective for subsequent discussion. The sequence chronologies described in Table 3.3-2 are for the low probability scenarios described in Appendix B and as follows:

- Successful scram, reactor isolated, loss of all coolant injection (TQUV)

- Successful scram, one SORV, loss of all coolant injection (TPQUV)

The decay heat level and the time available for operator action are critical to the analysis of event failure probabilities. Two aspects of the quantification of event failure probabilities which may be enhanced during recovery at low power operation; but for which only a minimal level of credit is taken in challenges from full power operation, are as follows:

1. <u>Operator action</u>: For low power operation, there may be a significant amount of time available for the initiation of contingency actions including connections of the fire water system to the ultimate core cooling connection, or using an onsite fire pumper truck as a backup to the pumps used for injection to ultimate core cooling connection (i.e., the diesel fire pump and/or the service water pumps).

2. <u>High pressure, low capacity system availability</u>: An additional plant system may be used for accident mitigation from an initial power level of 5%. The CRD pumps are considered a successful means of restoring and maintaining water level following shutdown. The high pressure capability of the CRD pumps makes them a valuable asset in accident mitigation.

TABLE 3.3-2
EVENT TIMING FOR CORE VULNERABLE SEQUENCES
(5% INITIAL POWER)

EVENT	TQUV FULL POWER	TQUV LOW POWER	TPQUV FULL POWER	TPQUV LOW POWER
Scram	0 min	0 min	0 min	0 min
Core Uncovery	30 min	18 hrs	30 min	30 min
Core Heatup	30 min	11-13 hrs	30 min	4 hrs

3.3.1.1 Initiator Frequency Quantification

This subsection provides the rationale used to requantify the dominant accident sequences involving any of the transient initiating events listed in Table 3.3-1. The quantitative input for this discussion is summarized in Table 3.3-3 which lists the estimated scaling factors for initiator frequencies. For additional perspective, refer to Section 3.5 of the Shoreham PRA for discussion of quantitative estimates that apply to challenges from full power operation.

The wear-in period for plant equipment (i.e., the initial 1 to 2 years of operation) has been shown by operating experience to result in a higher than average number of plant challenges. In this analysis, it is assumed that the number of challenges for most anticipated transients will be twice as high as the average for a mature plant. The one exception involves turbine trip initiators. As stated in Section 1, the main turbine will remain idle during much of this phase of operation. This reduces one of the causes of turbine trips, that of failures related to the turbine itself. Appendix A of the PRA also assigns recirculation pump failures and spurious instrumentation trips to the more general category of turbine trips, which are still applicable events for the plant configuration at low power operation. Therefore, while the first contributor is reduced, the others may occur with a higher than average frequency for the reason stated above. In the absence of data, it is judged that these effects are judged to offset each other, leaving no net change in the estimated frequency of turbine trips. However, to provide the Shoreham operating staff the flexibility to "spin" the turbine with protective trips in place, the evaluation includes a turbine trip challenge frequency of twice that of a mature plant.

For the category of low frequency initiating events, no significant impact on the point estimate frequencies can be found. If a trend does develop, it may be that mature plants will have significantly fewer initiating events of this type. These initiator frequencies are derived from relatively sparse experience data, for which there is no specific correlation to plant age. Therefore, they are considered as likely for a mature plant as they are during the initial stages of low power operation.

3.3.1.2 Mitigating System Conditional Failure Probability

The conditional failure probability of mitigating systems is subject to many of the same arguments used above for the initiator frequency. The exception however, is that the generic data from industry used to characterize these system failures already includes a large fraction of "wear-in" data. Therefore, the component failure probabilities are already judged to incorporate a substantial fraction of the potential for increased system unreliability. Nevertheless, in order to ensure that the "wear-in" period is adequately treated, an additional factor of two is included in the system reliabilities. This factor of two is also applied to common cause failures that may impact multiple systems. The following discussion summarizes the rationale used to develop scaling factors for various events.

Event C - Successful Scram, or, Criticality Control: A change of a factor of two in failure probability is assumed here for start-up operation at low power consistent with the above discussion.

Event M - SRVs Open as Required: A scram challenge at full power also causes (or is caused by) a turbine trip. The subsequent reduction in steam flow can sometimes lead to a brief pressure challenge in the reactor coolant system (RCS), forcing several relief valves to open. Such a pressure transient is not applicable for the assumed plant configuration in this report. As long as the MSIVs and turbine bypass valves remain open, no significant pressure rise in the RCS is expected following a scram. In cases where the reactor becomes isolated from the main condenser (e.g., MSIV closure) the pressure transient for low power operation may be less severe, and in this analysis, it is assumed that only one group of SRVs (3 of 4) is initially challenged. This is expected to substantially reduce the failure probability for this event. Due to the sparseness of available data, the significant redundancy in this function, and the minimal impact of failure of this event on subsequent calculations, no change in the failure probability is considered in this analysis.

Event P - SRVs Reclose, RPV Integrity Maintained and LOCAs not Induced: The key point to consider for this event is that a different failure criterion from that described in the PRA is used for this analysis. Because of the very low

decay heat levels following a scram from low power, this study assumes that one SORV is sufficient to rapidly depressurize the reactor. The significance of this event is that it accelerates the rate of inventory loss from the RPV similar to a LOCA, thereby reducing the time available for operator action.

A significant uncertainty in the accident sequence quantification is the number of challenges to the SRVs, which is roughly proportional to the calculated failure probability of this event using the constant failure rate model. For cases in which there is no vessel injection, the RPV heats up, pressurizes, and causes SRVs to lift; a process which is repeated a number of times until RPV inventory is depleted. Since the boiloff process is slow, it is expected to be many hours between subsequent SRV challenges. Therefore, in keeping with shutdown procedures, the most likely scenario involves the operator who is instructed to manually open SRVs to prevent cycling. A second point to consider is the initiation of the steam driven pumping systems, HPCI and RCIC. Because of the long response time available, it is most likely that these systems will be manually initiated to control RPV level as necessary, and also, because of the low decay heat levels, serve to depressurize the reactor without causing subsequent challenges to the SRVs. Therefore, only one demand on the first group of SRVs (3 of 4) is assumed. Despite the substantial reduction in the number of SRV demands, the failure probability of this event is governed by the revised failure criterion. Based on the failure rate data supplied in Appendix A of the PRA increased by a factor of two (to account for equipment wear-in), the failure probability of this event is calculated to be .02/challenge, which is an order of magnitude greater than that calculated in the PRA. This conditional failure probability is found to adequately quantify induced LOCA events following a transient initiator.

Event Q - Feedwater Availability: This event can be viewed more generally as "High Pressure RPV Injection from Non-Safety Systems". Within the Shoreham PRA following shutdown from high power, feedwater is the only system given credit for this function. For shutdowns from low reactor power and success at event P, however, the flow from the CRD pump becomes sufficient to maintain coolant inventory. This flow, which is normally available after a reactor scram, represents a significant improvement in the reliability of RPV injection. The potential link to the feedwater system is through the normal AC power system.

As shown in the transient event trees of the PRA, relatively high estimates of feedwater unavailability are dominated by operator errors and hardware problems which would be relatively independent of the CRD system. Therefore, the combined reliability of these two systems is estimated to be improved by the factor of 0.99/challenge to account for the CRD system flow.

Events U' and U" - RCIC and HPCI Availability: These turbine driven pumping systems are capable of operating at relatively low steam supply pressures but do have 2200 rpm minimum throttling speed limitations. Therefore, operation at very low decay heat levels may become erratic because continued operation of the pump may significantly reduce reactor pressure. It is judged that the more likely use of these systems will be for intermittent RPV level control. These conditions are assumed to increase the potential for hardware, valve and for control failures and therefore, the combined unavailability of these systems is increased by a factor of 2 for this analysis. Coupled with the increase due to component "wear-in" this results in a factor of four increase in combined system unreliability.

How does this affect battery lifetime?

Event X-Timely Reactor Depressurization: As stated for Event P, one open SRV is considered adequate for reducing reactor pressure. Although this would appear to reduce the demand on the ADS, it is noteworthy to recall that the PRA evaluated the significant redundancy within the SRV depressurization system and identified common cause failures as the dominant contributors. The common cause conditional failure probability is increased by a factor of two to reflect the "wear-in" period. However, since the low initial power cases afford the operator a substantial amount of time to implement corrective action it is judged that the following three options would provide the operator the capability to depressurize over the extended time available for the low power initiators:

- o HPCI steam line with discharge to the suppression pool
- o RCIC steam line with discharge to the suppression pool
- o MSIV or MSIV drain line to the main condenser

Using these options the conditional failure probability is reduced by a factor (five). The combination of the above factors is a net reduction in the

conditional failure probability of depressurization of a factor of 2.5. For cases with an SORV there would not be a requirement for the depressurization function because of the relatively rapid pressure reduction.

Events V', V", and V''' - Core Spray, LPCI, and Condensate System Availability:

For shutdowns from low initial power levels, the extended period of time available prior to core uncovery along with the reduced makeup flow requirement allow for possible contingency actions to be initiated. Among these contingencies is the use of service water pumps and the diesel fire pump injection flow path via the ultimate core cooling connection. Given that the sequence does not involve an early loss of core inventory (e.g., an SORV) the failure probability of the combined event (low pressure injection availability) is estimated to be a factor of 5 lower. This quantitative assessment is based on the likelihood that at long times, i.e., greater than 4 hours the technical support center (TSC) would be manned and creative decisions such as the use of service water could be implemented. However, since the diesel fire pump procedure for injection has not yet been implemented, this contribution to improved system reliability is not assessed at a high success rate. If the sequence timing is accelerated by an SORV, then the combined failure probability is estimated to be only a factor of 2 lower due to the time constraints on operator action. (The estimate of .5 in unreliability is judged to be conservative).

Events W', Z, and W" - Containment Heat Removal: As discussed in the introduction of Section 3.3, the time frame for core vulnerable sequences following loss of containment heat removal is on the order of several days. Plant failures under these conditions are not considered to be credible contributors to risk, therefore, a detailed quantification of dominant accident sequence frequency was not carried out.

3.3.1.3 Dominant Accident Sequence Quantification

Using the scaling factors derived in this subsection, a subset of the Class I dominant accident sequences is requantified in Table 3.3-4 to estimate the frequencies of equivalent sequences initiated from low power operation. All

Transient induced LOCAs are significant in that they tend to be more dominant probabilistically than small or medium LOCAs. From a plant response perspective, these sequences are important since the rate of coolant lost through the safety relief valve could lead to rapid core uncovering. If coolant inventory is not reestablished, the core could heat up and core damage could occur early.

Following full power operation, the decay heat rate is sufficiently high that the reactor pressure could stabilize above the minimum HPCI and/or RCIC operational range, therefore, continued coolant injection by the HPCI and/or RCIC pumps is possible.

For low power operation, while the decay heat rate is low, the reactor operating pressure is nominally the same as for full power operation. Since steam discharge through the SORV is dependent upon the pressure, the rate of coolant loss for a given RPV pressure in both cases are equivalent.

A negative effect of the low decay heat rate for low power operation is that the reactor pressure may not be maintained above the minimum operating range of the steam driven HPCI and RCIC pumps given a SORV. Therefore, these pumps may become unavailable much sooner for LOSP SORV cases initiated at low power than for full power operation. The positive impact of the low decay heat rate, however, outweighs the negative effect in that a substantial reduction in the coolant makeup capacity is required to maintain sufficient core cooling.

In this analysis, several cases of Category 2 transient events were investigated:

(1) Stuck open relief valve (SORV) where the RPV coolant inventory lost through the SORV is not recovered

(2) SORV where coolant makeup through the high pressure injection systems is available initially.

(3) Transient events with immediate blowdown through the ADS valves, without coolant makeup. This case is judged to encompass large LOCA sequences in the steam lines.

(4) Transient events with controlled blowdown through the SRV, at a rate of less than 100°F/hour. The blowdown is initiated at 30 minutes in accordance to a procedural limit on drywell conditions.

For those sequences where inventory lost through the SORV is assumed to be replenished, the plant models considered that the coolant injection is initially provided by the HPCI and/or RCIC pumps during depressurization (while the reactor pressure is above the required set points). These sequences were further examined with minimal coolant make-up. It was assumed that RPV inventory was subsequently maintained by the CRD pumps in the long term following reactor depressurization and the single cycle of HPCI injection.

For this category of transient events, the reactor depressurization would leave the core uncovered. Without coolant makeup, the core would then heat up. Due to some steam cooling during the depressurization stage, radiative heat transfer and low decay heat levels, core vulnerability is estimated not to occur until after 2 to 7 hours following the transient event at five percent power operation. This would be extended further if the steam driven pumps are given credit and coolant injection occurs prior to reactor depressurization below the HPCI pump operational range.

Case 1: SORV Without Coolant Makeup

The particular accident sequence evaluated here assumed the reactor is shutdown and isolated following the initiating event. The RCS coolant temperature increases and the reactor pressure approaches the SRV setpoint of 1130 psia. The SRVs open to relieve reactor pressure and one SRV fails to close. Due to the low power levels of the core, the initially lower RPV pressure, and the initially lower core average temperature, the RCS coolant heatup to the SRV setpoint does not occur almost immediately as would be expected during the same transient event initiated at full power. In this analysis, it is estimated that the SRV setpoint would be reached

after approximately 20 minutes following the initiating event. The MARCH analysis predicts core uncovery occurring at 25 to 30 minutes, and core heatup is initiated with the core fully uncovered. Core overheat occurs in a steam starved environment. The reactor is depressurized and the coolant level is below the bottom active fuel height. These factors lead to minimal steam generation rates, thus cladding oxidation does not contribute significantly to core overheating. The estimated time to reach core vulnerability is approximately 4 hours following the initiating event for this case of Category 2 accident sequences.

Case 2: SORV With Coolant Makeup

This case of transient induced LOCA sequences involves a similar transient event as Case 1 described above. In Case 2, following the SORV, during the reactor depressurization, the HPCI pump is assumed to be activated automatically. Because of the coolant loss due to the combined effect of flashing and boil off, it is assumed that water level in the reactor does not exceed Level 8 thus the HPCI pumps are not tripped prematurely. It is further assumed that the HPCI pumps would continue to operate as long as the reactor pressure is above 150 psig. Therefore, the boundary conditions for coolant boil-off would be a reactor at 150 psig, the water level at approximately NWL, and a stuck open relief valve. Reactor depressurization continues until the reactor pressure drops to approximately 20 psia. Boil-off of the remaining coolant inventory at 20 psia involves a longer time than would occur at an elevated pressure due to the higher heat of vaporization at the lower pressure compared to 1130 psia.

In this scenario, following the initial core recovery, without subsequent coolant makeup from other modes of coolant injection, it was determined that coolant boil off to the top of active fuel would occur within 20-25 hours. Subsequent core vulnerability is not expected to occur until after 12 to 15 hours. Therefore, with a more realistic assessment of SORV sequences for LOSP cases, there is a potential for extending the time

before core vulnerability occurs. Since there exists a driving force for the HPCI/RCIC pumps (i.e., sufficient steam pressure and flow rate from the RPV during reactor depressurization), it is judged that there is sufficient time to recover the other modes of coolant injection. The CRD pumps providing subsequent coolant flow into the vessel was found to mitigate this accident event.

Case 3: Transient Events with Immediate Blowdown

Case 3 is the limiting scenario of the Category 2 accident events involving transient induced LOCA accident sequences. This scenario is intended to encompass large LOCA accident sequences as well, in which the reactor is depressurized to containment conditions almost immediately. As in the SORV cases, the core is completely uncovered after blowdown. But since blowdown occurs very rapidly, the stored thermal energy in the core is not sufficiently dissipated prior to initiation of core overheat. In this sequence, the core becomes vulnerable within 2-1/2 to 3 hours, following the initiating event.

Case 4: Transient Event with Controlled Blowdown

This scenario considers a controlled depressurization of the reactor initiated by the operator to meet procedural limits of drywell temperature. During loss of all AC power sequences, the drywell coolers become unavailable, and the drywell atmosphere could heat up beyond 296°F which would require reactor depressurization. The drywell temperature climbs rapidly during the first 5 to 10 minutes before heat transfer to the drywell liner is established. In this analysis, the drywell temperature limit is assumed to be reached within 30 minutes at which point the reactor is depressurized at a rate not to exceed 100°F per hour. This scenario effectively extends initiation of core heatup due to the concommitant steam cooling during the depressurization stage. Although core uncovery occurs within a few minutes, the blowdown to a pressure of

150 psig* is reached at approximately 150 minutes following the initiating event. Core heatup subsequently follows and core vulnerability could occur after approximately 6-7 hours.

B.3.3 Category 3 - Anticipated Transients Without Scram

This category of accident sequences include those low frequency event sequences in which an anticipated transient coupled with failure to insert the control rods may occur. In the Shoreham PRA, the evaluation of postulated ATWS accident events indicated that these sequences cou'd potentially lead to a more severe containment challenge compared with the other types of accident sequences investigated thus far. Following operation at 100% power, pool heat-up and containment overpressure occurs rapidly. The estimated time of core vulnerability determined in the PRA was approximately 30-40 minutes.

For ATWS events for which the condenser is isolated, the initial low power level of 5 percent results in a slower rate of suppression pool heat up which then provides more time for the operator to take action in mitigating the accident. In this evaluation, it is estimated that several hours would be required to heat up the suppression pool to saturation and several more hours for the containment to reach its ultimate pressure capacity. This assumes that the reactor power drops to approximately three percent of rated power or 60 percent of the initial low power level of five percent. RCS coolant inventory makeup is derived from the turbine driven HPCI pumps, and both trains of the residual heat removal heat exchangers are postulated to be unavailable. The significance of high suppression pool temperature relates to the operability of the HPCI pumps under adverse conditions, e.g., lube oil

* The emergency procedure guidelines call for the operator maintaining the reactor pressure at 100-150 psig. It is assumed in this evaluation that the operator would tend to keep the reactor pressure from dropping well below 150 psig. This would allow sufficient driving force for the steam turbine HPCI pump driver thereby ensuring availability of the HPCI system.

degradation. For some ATWS sequences evaluated in the PRA, continued coolant injection may jeopardize containment integrity which in turn could result in the degradation of the ECC systems and inability to maintain coolant inventory. Therefore, two cases of ATWS isolation events were considered in this evaluation:

(1) Coolant injection is terminated at the point when the suppression pool temperature reaches 240°F. In this scenario, coolant boil off would occur in an intact containment.

(2) Continued coolant injection is assumed, leading to containment failure and ECC failure. This is subsequently followed by boil off of RCS coolant inventory in a failed containment.

In both cases described above, core vulnerability would follow shortly after core uncovery and dryout at decay heat energy levels.

The calculations performed predict an estimated time of approximately 3-4 hours to heat up the suppression pool to 240°F. For Case 1, subsequent to coolant injection being terminated at this point, the RPV inventory to the TAF is boiled off with the reactor at 3 percent of rated power. The core is uncovered within 30 minutes after termination of coolant injection and heats up at the fission product decay heat power. It is estimated that core vulnerability would occur at greater than 5-6 hours following core uncovery. For Case 2, assuming that coolant injection is maintained, the containment is estimated to fail by overpressure after approximately 6-7 hours. Following containment failure, ECCS injection is terminated. Coolant boil off is estimated to uncover the core after 30 minutes and core overheat would subsequently occur after approximately 5-6 hours. In this analysis, the energy released via the steam flow to the Terry turbine HPCI/RCIC pump driver was not considered. It is estimated that pool heatup and containment failure could be delayed by approximately 5-10 percent if this additional heat sink is considered in the analysis.

Transient events with failure to shutdown the reactor is potentially a more severe accident category because of the higher thermal energy release rate compared to shutdown conditions. This category of accident events result in

the SRVs being open more often than would be if the reactor were shutdown. The cycling of the SRVs present a challenge that could lead to one of the SRVs failing to close. An ATWS event coupled with a SORV was investigated to determine the impact of the stuck open relief valve on plant response during the ATWS event. It was determined that the reactor pressure could stabilize at approximately 350 to 400 psia for this sequence. The impact on sequence timing would not be noticeably different. However, a significant perturbation would be the reactor coolant loss following termination of injection flow. Blowdown could occur from 350 psi to containment conditions due to the stuck open relief valve following decay in the core power which could result in a more rapid core uncovering. If the core becomes fully uncovered, a steam starved core condition would be expected during the core heat up phase delaying the core heat up period to some extent due to reduced clad oxidation.

B.4 Sequence Perturbations

The three categories of accident sequences described above were also examined to determine the sensitivity of the accident event timing for power levels less than the reference value of five percent. In this sensitivity evaluation, the reactor was assumed to be operated at a constant level of 2.5 percent of rated power; not the more likely scenario involving impulse power fluctuations below the reference five percent level. The accident progression determinations were conducted consistent with the five percent power accident analysis.

Because of the decaying nature of fission product energy release rate, the time scales of the accident sequences following shutdown are not always inversely related to the initial power level, i.e., at 2.5 percent power, the time to core vulnerability is not exactly twice that of the 5 percent power level case. In general, the required time to boil off the same amount of water inventory or heat up of the fuel would vary depending upon the time from shutdown at which boil off or core heat up is initiated. This is apparent from the results of the assessment of the sequence perturbations of the

category 1 transient events (isolation without makeup) initiated at five percent power level described in Section B.3.1.

The results of this sensitivity evaluation indicate that for the transient events and isolation cases without coolant makeup, (event category 1 described above) the time to initial core uncovering estimated for the 2.5 percent power level case is approximately 45 to 50 hours, and the subsequent core vulnerability is estimated to occur after another 26 to 30 hours. If the reactor inventory is assumed recovered after boil off to Level 2, the time to core vulnerability would be found extended accordingly. This assumes only a single cycle of HPCI and that subsequent coolant injection does not occur. Event category 2 evaluation (transient induced LOCAs) shows the same trend, and core overheat is predicted to occur at greater than 10 hours.

For the ATWS scenarios, for which no containment heat removal capability is assumed, containment integrity is jeopardized and coolant injection is lost at 6-7 hours. This is followed by boil off and core overheat occurring after another 8 hours after containment failure. For this sequence involving failure to bring the reactor subcritical, the initial phase of the accident assumes that the reactor is at 60 percent of initial power level. Therefore, the rate of suppression pool heatup and containment pressurization is expected to be directly related to the assumed power level.

B.5 Summary and Conclusions

The accident analysis performed for the potentially risk dominant sequences described above indicates that a significant risk reduction (in terms of estimated frequencies) during low power operation is possible. The substantial time required to reach core vulnerability for each event category discussed in Section B.3, range from 2-3 hours to several days. This range of times would most likely provide sufficient time for operator action to mitigate the

accident. In addition, the required mitigative capacity of coolant injection sources is significantly reduced such that other coolant injection success paths are possible.

The results of this evaluation as summarized in Table 3 provides an indication of the time windows available for the operators to implement mitigative actions. This table also shows the range of times prior to core vulnerability for the accident sequences studied in this appendix given some perturbations in the time delay prior to initiation of coolant boil off. It appears that at low power levels, the time scales of the accident sequence progression to the point where the core or containment integrity may be lost are quite long that substantial times are available for operator action to mitigate the accident.

From an overall risk perspective (i.e., frequency and consequence considerations), the potential off-site public health impact of these accident sequences initiated at low power would be reduced significantly. Because of the low decay heat energy release rate, the containment integrity will likely be maintained for several days given that the accident does proceed unchecked to a core meltdown. This will undoubtedly remove significant portions of the airborne radionuclide materials from containment, thus substantially reducing the amounts of fission products that could be released to the environment. On the basis of this assessment, and considering the general aspects of radionuclide behavior, it is concluded that significant reduction in the source terms is possible at low power operation. It is estimated that for the extended times prior to containment failure and radionuclide release, a source term reduction factor ranging from 10 to 100 may be possible for similar accident sequences involving early containment failures studied in the PRA. Furthermore, the fission product inventory of the reactor core operated at low power restricted to five percent is a factor of 20 less than that at full power.

Table 3
SUMMARY OF EVENT TIMING FOR SELECTED ACCIDENT SEQUENCES
AT LOW POWER OPERATION

ACCIDENT CATEGORY	TIME DELAY PRIOR TO BOIL OFF	TIME TO CORE UNCOVERY FROM BOIL OFF INITIATION	TIME TO CORE OVERHEAT FROM CORE UNCOVERY	TOTAL TIME TO CORE VULNERABILITY
CATEGORY 1				
Isolation	0	18 hours	11-13 hours	30 hours
Transients	4 hours	29 hours	12-14 hours	2 days
	10 hours	33 hours	13-15 hours	2-1/2 days
	30 hours	35 hours	14-18 hours	3-1/2 days
CATEGORY 2				
SORV w/o makeup	0	25-30 minutes	3-1/2 hours	4 hours
SORV w/initial makeup	30 minutes	20-25 hours	12-14 hours	1-1/2 days
ADS (LOCA)	0	0	2-3 hours	3 hours
Controlled Blowdown	0	45-50 minutes	6-7 hours	7.5 hours
CATEGORY 3				
ATWS with containment intact	3-4 hours	30 minutes	5-6 hours	8-10 hours
ATWS with containment failed	6-7 hours	30 minutes	5-6 hours	11-13* hours

* Time from initiating event given continued coolant injection from the suppression pool, i.e., the impact of high pool water temperature (greater than 240°F) on HPCI/RCIC operability is not considered in the analysis.

This analysis focused on LOSP cases for which the plant systems considered for coolant injection did not include those requiring off-site power. Extrapolating the results of this analysis to other accident initiators, could likewise lead to extended times available for operator action for similar accident sequence progression. In summary, a significant reduction of risk to the public (conservatively estimated as at least a factor of 20 to possibly 200 due to the reduced potential off-site consequences alone) is judged likely for the spectrum of accident events analyzed in the PRA for low power operation at Shoreham.

REFERENCES

1. Shoreham Nuclear Power Station, Probabilistic Risk Assessment, SAI-372-83-PA-01, June 1983

2. Draft Report on Status of Validation of the MARCH-2 Computer Code. Battelle Columbus Laboratories, Columbus, Ohio, July 11, 1983

3. Division of Systems and Reliability Research, Office of Nuclear Regulatory Research, Battelle Columbus Laboratories, MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual, R.O. Wooten, H.I. Avci, NUREG/CR-1711 BMI-2064, Columbus, Ohio, October 1980

4. LILCO Interoffice Memo, NFD-83-016, January 24, 1983

5. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, August 29, 1979

6. Personal Communication, R.J. Paccione (LILCO) and Z.T. Mendoza (SAI) dated March 22, 1984

APPENDIX C

SNPS/LILCO GRID ELECTRIC POWER SYSTEM DESCRIPTION

ere are two off-site power sources at SNPS which are physically and ectrically independent [1]. The primary source of off-site power to the int is via the Normal Station Service (NSS) transformer which is connected ween the SNPS generator circuit breaker and the 138KV switchyard. The ondary source is through the Reserve Station Service (RSS) transformer ich is connected to the 69KV transmission system. A schematic diagram of e 138KV transmission system in the area surrounding SNPS is shown in Figure 1. A one line diagram of the main portion of the SNPS electric power system shown in Figure C-2. The 138KV and 69KV transmission lines from the plant tend out to various substations at nearby locations on the LILCO grid. One the grid connections, the Holbrook Substation, currently has a connection a gas turbine generator with black start capability.

e on-site gas turbine generator considered for the black start modification SNPS is connected to the 69KV system on the site near the RSS transformer. e configuration of the 69KV system with respect to the on-site gas turbine shown in Figure C-3.

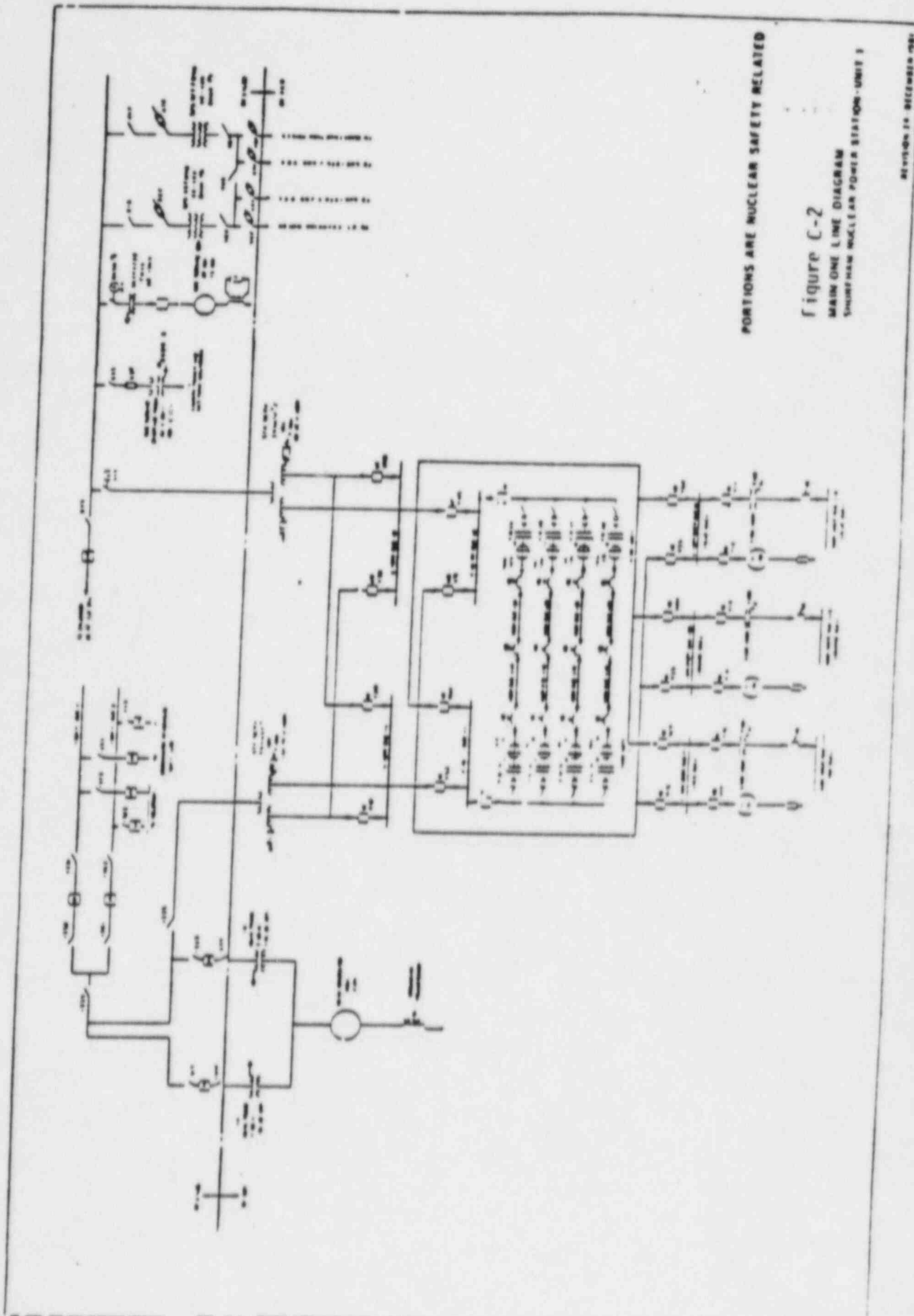


Figure C-2
MAIN ONE LINE DIAGRAM
SHOREHAM NUCLEAR POWER STATION-UNIT 1

JRH1

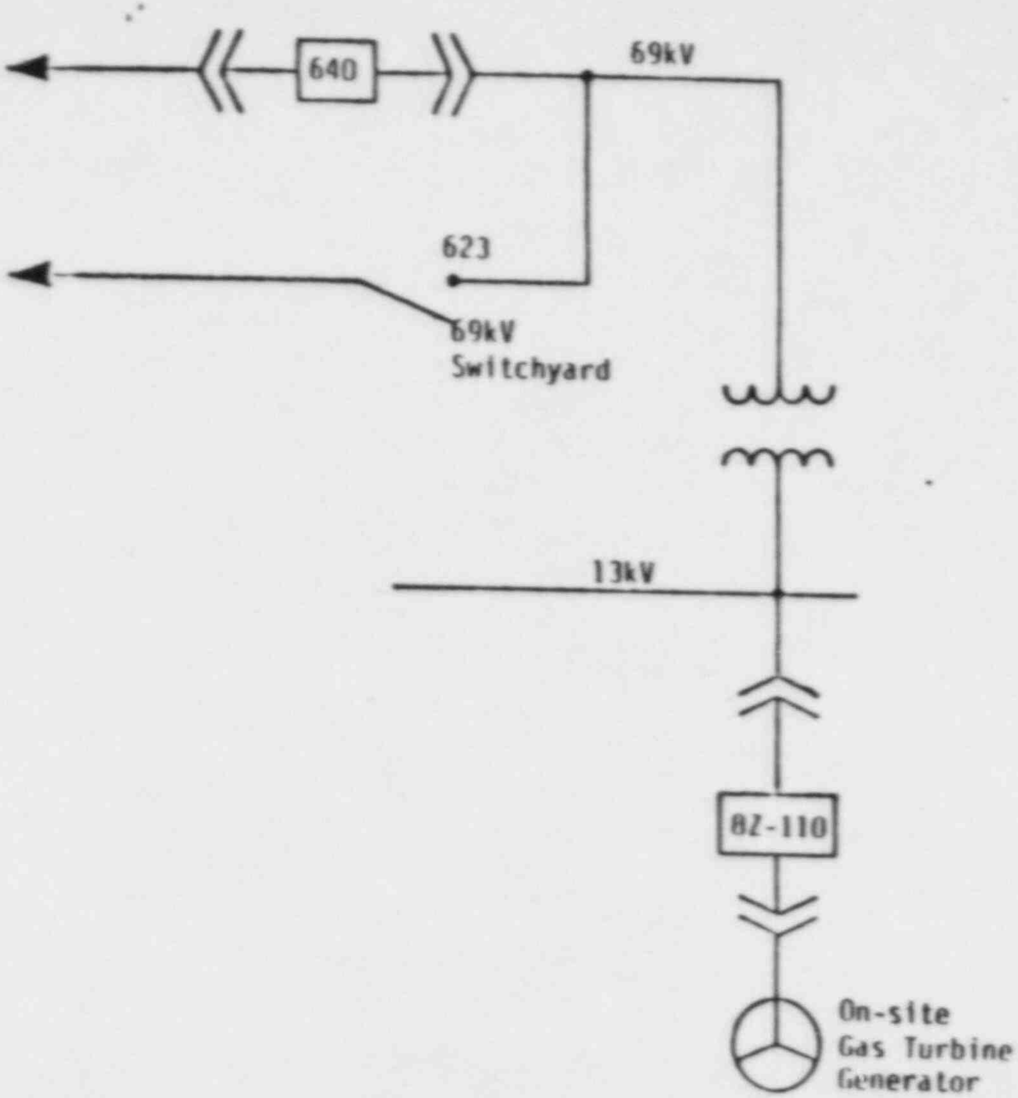
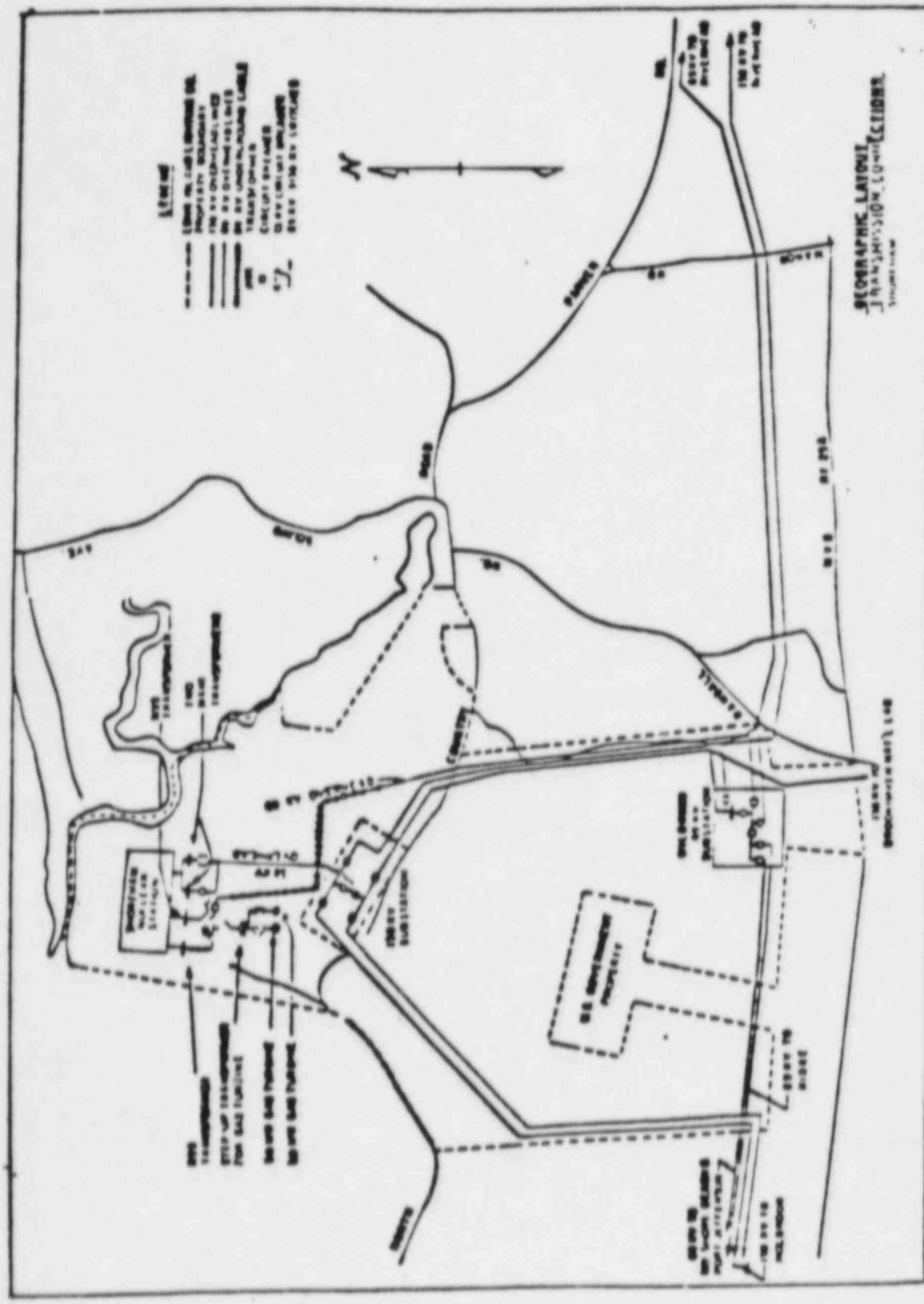


Figure C-3 Schematic Diagram of Shoreham 69KV Switchyard



REFERENCES

1. Shoreham Nuclear Power Station, Final Safety Analysis Report, Docket No. 50-322.

2. Letter from R.J. Paccione/R.S. Zambratto (LILCO) to E.T. Burns (WLA), April 9, 1983.

APPENDIX D

ASSESSMENT OF LOSP EVENT DATA AND APPLICATION OF DOMINANT ACCIDENT SEQUENCES

Two of the key parameters used in the Shoreham PRA for estimating the frequency of core vulnerable conditions following LOSP are: (1) the frequency of LOSP events, and (2) the time required for recovery of off-site power. These parameters are derived from the LILCO grid reliability data base and EPRI NP-2301, respectively. In the latter, data from a nationwide survey are examined and conditional probabilities of recovery events are estimated for Shoreham. The proposed black start modification provides a redundant and diverse method for restoration of off-site power.

The time required for recovery of off-site power depends heavily on the particular failure mode leading to the LOSP event. A review of the causes of the LOSP events in EPRI NP-2301 leads to the definition of three general categories of LOSP events: 1) grid failures or transmission line failures, 2) main switchyard failures, and 3) failures of both switchyards. These failure event categories allow the effectiveness of the black start gas turbine system at Shoreham to be modeled and a sensitivity of its effectiveness developed. The LOSP categories are described as follows:

Grid or Transmission Line Failures: Based upon a review of historical data throughout the U.S. [1,2], grid failures range from a total system shutdown to relatively minor switching errors that de-energize substations. With proper switching, either the black start system feeding the Holbrook Substation (22 miles from Shoreham) or the Shoreham on-site gas turbine with black start could restore power to Shoreham.

Transmission line failures are generally weather induced by causes such as wind storms or ice storms resulting in failures of several transmission lines. While the transmission lines at Shoreham are generally widely separated, the

69KV circuit does share the same right-of-way as one of the two 138KV circuits for a distance of approximately one mile. Therefore in cases in which the transmission lines become unavailable, the Holbrook black start system may be ineffective while the Shoreham on-site gas turbine could still provide power to the Shoreham on-site electrical distribution system.

In this evaluation it is judged that the likelihood of recovery from grid failures is similar for LILCO and for other grids in the U.S. In other words, the black start gas turbine capability at Holbrook is judged to be adequately included in the operating experience data base which reflects the strong possibility of recovery from such events. Therefore, in the assessment of the on-site black start gas turbine capability, both grid failures and transmission line failures can be lumped together for the purposes of quantification.

Main Switchyard Failures: These events apply to the range of failures which could occur in the vicinity of the main switchyard. These failures are considered to be independent of the availability of the 69KV system. In these cases, the 69KV system is expected to remain energized, so there is no substantial advantage to the addition of the on-site black start gas turbine.

Failures of Both Switchyards: These events involve common mode failures between the two switchyards. In these cases, the black start capability would not be effective since the 69KV switchyard is required to direct the power from the on-site gas turbine into the normal 4160V buses.

The allocation of the failure modes in EPRI NP-2301 into these three categories is crucial in the assessment of the relative worth of the on-site gas turbine and its potential public safety improvement. This allocation turns out to be one of the principal contributors to the uncertainty associated with the calculated reduction in core vulnerable frequency associated with the Shoreham design modification. Therefore, the following quantification is structured to provide a sensitivity on the best estimate values to indicate the potential variation based upon data uncertainty.

Table D-1 shows the categorization of the LOSP initiated events in EPRI NP-2301. Table D-1 has been constructed with some subjective judgement used to characterize the failure modes. In particular, the failure modes with potential impact on redundant switchyards have been inferred from the data; that is, those failure modes which are judged possible to cause a simultaneous failure of both Shoreham switchyards are identified. The conditional probability for failure to recover off-site power for each time phase is obtained from the sum of failures for each of the LOSP event categories. EPRI NP-2301 contains a large amount of useful data to provide an overview of what an "average" plant might look like. However, one must be prudent in the application of these data on a plant specific basis. In particular, associating the Shoreham-specific LOSP frequency with the generic failure modes from EPRI NP-2301 may underestimate the benefit of the black start capability.

Information presented in each column of Table D-1 is described below:

(1) Nuclear plant at which the LOSP is recorded.

(2) Incidents of LOSP which are caused by either total grid blackouts or transmission line failures [0 = less than 30 minute duration; x = is greater than 30 minute duration].

(3) Incidents of LOSP in which a main switchyard failure is involved.

(4) Incidents of LOSP recorded in column (3) which also involve multiple switchyard failures or the potential for such are identified in column (4). The probabilistic sensitivity analysis of Shoreham results in the calculation of two cases:

A) Main switchyard failure considered to have a high likelihood of impacting the redundant switchyard.

B) Main switchyard failure considered to have a high or uncertain likelihood of impacting the redundant switchyard.

(5) The duration of each LOSP event.

(6) The number of grid connections at each plant.

Table D-1

CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Beaver Valley		0			:17	2
Calvert Cliffs		X		X	5:29	2
Davis Besse		X*		X*	----	3
"		0			:26	3
Dresden 1	X				25:40	9
Farley		X		X	4:59	2
Fitzpatrick		0			<:01	2
"		0			:03	2
Fort Calhoun		X			11:05	7
"		X		X	:54	7
"		0			<:01	7
Ginna		0			:30	2
Ginna	X				:40	2
Haddam		0			:29	2
"		0			:09	2
"		0			<:01	2
"	0				:20	2
"		0			:16	2
Humbolt Bay		X*			----	2
Indian Point	X				6:28	2
La Crosse	0				:14	2
"		X			1:01	2
"		0			:20	2
"		0			:02	2
"		X		X	1:50	2
"		0			:10	2
Millstone 1		X		X	5:35	2
Nine Mile Point		X	X	X	24:37	2
Oconee		0			<:01	2
"		X			1:00	2
Oyster Creek		X*	X*	X*	----	3

See notes on following page.

Table D-1 (Continued)
CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Palisades		X			:56	5
"		X		X	4:45	5
"		X		X	3:30	5
"		X		X	1:30	5
Pilgrim	X				2:40	3
"	X				8:54	3
Point Beach		O			:08	2
"		X			:55	2
Quad Cities		X			1:11	4
San Onofre		X	X	X	4:59	7
"		O			:04	7
"		O			<:01	7
St. Lucie	O				:08	2
Yankee Rowe	X				:37	2
The total number of LOSP events is 45						

X - Indicates an event lasting >30 minutes.
O - Indicates an event lasting <30 minutes.
* - Assumed, based on the specific failure mode.
Included in the calculations as a failure lasting >4 hours.

The recovery factors determined for LOSP events in this analysis incorporate some subtle but potentially important distinctions regarding the source of the data, i.e., the data set is composed of a spectrum of failures from rather minor single failure events through severe weather conditions affecting multiple transmission lines. Based upon the data in EPRI NP-2301 (causes and durations), it is clear that there is a strong coupling between the duration of the loss of off-site power outage and the particular failure mode. However, in the assessment of the on-site gas turbine capability the use of the data to allocate exact failure modes extends the statistical significance of the data to its limits. For this reason, this analysis attempts to examine the general trend of the coupling of the failure modes (e.g. weather) and

duration of LOSP and to provide both optimistic (CASE A) and pessismistic (CASE B) assumptions in interpretation of the data to account for uncertainties in classification o event. From this perspective, long duration transmission line failures take on an increasing importance. The analysis incorporates these data directly into the probabilistic evaluation including the strong coupling between failure mode and duration of LOSP.

Table D-2 summarizes the data and classifies the failure modes by time phase consistent with the PRA. The information in the table may be used to derive estimates for recovery values used in the recovery logic model.

The time phases which have the highest contribution to core vulnerable frequency following LOSP initiators, i.e., time phases III and IV, have a substantial fraction of events for which off-site power has not been recovered based upon historical nuclear plant operating experience.

Table D-2

SUMMARY OF LOSP EVENTS BY TIME PHASE

TIME PHASE	GRID AND OFF-SITE TRANSMISSION	MAIN SWITCHYARD	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS	
			CASE A (optimistic)	CASE B (pessimistic)
PHASE I > 30 MIN	6	19	3	12
PHASE II > 2 HR	4	11	3	9
PHASE III > 4 HR	3	10	3	8
PHASE IV > 10 HR	1	2	1	1

REFERENCES

1. Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301, March, 1982.

2. Scholl, R.F., "Loss of Off-Site Power Survey Status Repor", Revision 3, Report of the Systematic Evaluation of Program Brench, Division of Licensing, U.S. NRC.

APPENDIX E

SENSITIVITY STUDY OF THE CORE VULNERABLE FREQUENCY ASSOCIATED WITH LOW POWER TESTING

The Main Report provides an indication of the assumed plant configuration for low power testing. For simplification, steady state operation at 5% power is assumed. In reality, however, the plant will be in a state of flux as numerous system tests, inspections, and measurements take place. The assumption of a steady state power level of 5% is judged to be a conservative approach to the assessment of risk for low power operation. The purpose of this Appendix is to provide additional bases for concluding that this approach is conservative. Specifically, a summary discussion of the detailed operational aspects of low power testing [1] is provided along with a best estimate, rather than conservative upper bound, evaluation of the core vulnerable frequency.

E.1 DETAILED OPERATIONAL ASPECTS OF LOW POWER TESTING

The initial plant testing at Shoreham involves a series of four phases defined as follows:

Phase I:	Fuel loading and precriticality testing
Phase II:	Cold criticality testing
Phase III:	Heatup and low power testing to rated pressure/temperature conditions (approximately 1% rated power)
Phase IV:	Low power testing (1-5% rated power)

The structure of the testing program is such that the plant must fulfill specific testing and operational objectives in each phase before continuing to the next. Component "wear-in" failures will be repaired as they arise, which implies that the plant will have a full complement of safety systems available for each new power level.

In this evaluation, it is judged that essentially no measurable risk can be associated with the first two phases of operation. This conclusion is based on two observations:

1. WASH-1400 [2] identified spent fuel handling accidents as a potential radionuclide release mechanism. Damage to unirradiated fuel would not fall into this category, however. Fresh fuel bundles can be handled in open air and would not be subject to melting or significant radionuclide release.

2. Cold criticality testing produces a negligible amount of heat (0.0001% to 0.001% of rated thermal power, or a maximum of 24kw). Under these conditions, there is essentially no need for RPV inventory makeup systems. If the RPV is inadvertantly drained, the core will become subcritical. Therefore, postulated precursors to core damage (e.g., LOCA or failure to scram) would have a negligible impact on core integrity.

Thus, Phases III and IV, in which the RPV is pressurized, steam is generated, and coolant makeup systems are required is judged to be the region of operation in which the first measurable risk due to radionuclide release appears. The testing program for these phases consists of repeated gradual heatups to a maximum of 5% full power followed by controlled cooldowns. Only a small fraction of the total operating period is spent near 5% power. Because of these power cycles, it is judged that decay heat levels, which are dependent upon the power history, can best be modeled using an intermediate valve between 0% and 5% and assuming steady state operation. For this reason, the core vulnerable frequency associated with operation at 2.5% power is estimated in the following section.

E.2 CORE VULNERABLE FREQUENCY DUE TO STEADY STATE OPERATION AT 2.5% POWER

As with the analysis presented in the Main Report, phenomenological calculations of postulated accident progressions are an essential starting point for assessing success criteria and quantifying event probabilities. The analysis presented in this section is intended to (1) review the results of MARCH calculations for accidents initiated for 2.5% power (2) identify and reassess events which are significantly different in timing or magnitude from the 5% power case; and (3) quantify these differences to estimate the core vulnerable frequency. This approach is judged to be the most effective method of

evaluating the sensitivity of the assumed power level. The results will be directly comparable to accident frequency estimates derived in the Main Report (5% power) and in the PRA (100% power).

E.2.1 ACCIDENT PROCESS CALCULATIONS

Appendix B presents the results of process calculations from MARCH for accidents postulated during low power operation. The emphasis is primarily on the evaluation of parameters assumed for steady state operation at 5% power. In particular, the differences between 5% and full power include an upwardly skewed axial power profile, an average core temperature of 640°F (representing a reduction of 400°F), and a reduced reactor pressure of 950 psia. Given these differences, the key modeling difference between the 5% and 2.5% power level calculations is the reduction in initial and decay heat power levels. With this in mind, Table 1 presents a summary of the estimated timing of postulated accident sequences in which the core becomes vulnerable to melting.

TABLE E-1
ACCIDENT SEQUENCE TIMING SUMMARY

SEQUENCE	TOTAL TIME TO CORE VULNERABILITY (HRS.)	
	5% POWER	2.5% POWER
Scram, Isolation, Failure of Coolant Injection	30	71 - 80
Scram, Large LOCA, Failure of Coolant Injection	3	10
ATWS, Containment Initially Intact	8 - 10	14 - 15

These timing estimates are judged to have a significant impact on the plant system success criteria beyond that already included in the Main Report. Three principal effects are as follows:

1. CRD pump flow is considered a viable alternative coolant injection source in accidents accelerated by a loss of coolant inventory. This applies to SORV cases, medium LOCAs, and large LOCAs, but not RPV ruptures.

2. RCIC alone is a viable alternative for coolant injection during an ATWS (at 5% power, a combination of RCIC and CRD flow is judged necessary).

3. The reactor power level following ATWS with RPT is estimated to be within the capacity of 1 RHR heat exchanger. This is based on the assumption that the power level will decrease by approximately 40% following RPT. Even if a single loop of RHR failed to match the ATWS power level, the challenge to containment is expected to be substantially extended. In these instances, a rationale similar to that described in the Main Report for dismissal of Class II challenges is judged applicable, i.e., the very long period of time available prior to containment failure represents a risk below that which can be credibly quantified. Therefore, such cases are judged to have a negligible frequency.

These success criteria are incorporated into the quantification of core vulnerable accident sequence frequencies in the remainder of this section. Additionally, revised event success criteria due to the timing estimates will be discussed as they arise.

E.2.2 QUANTIFICATION OF ACCIDENT SEQUENCE FREQUENCIES

This section corresponds to Section 3 of the Main Report. As such, a discussion and quantification of each of the four initiator types is presented; based on the revised success criteria for operation at 2.5% power.

E.2.1.1 LOSS OF OFFSITE POWER INITIATOR

The LOSP tree shown in Figure 3.1 of the Main Report consists of an initial subtree used to define groups of sequences with similar timing, followed by subsequent subtrees used for modeling time dependent events. For the reassessment at 2.5% power, the only event requantified is Event R: Recovery of Offsite Power. Table 3-1 presents a comparison of the estimated recovery probabilities for the 5% and 2.5% power cases. This requantification results in a reduced core vulnerable frequency estimate of 7.7E-7 events/reactor-year for LOSP initiators.

TABLE 3-1 CONDITIONAL PROBABILITY OF RECOVERY OF OFFSITE POWER AS A TIME DEPENDENT FUNCTION

ACCIDENT SEQUENCES	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (5% POWER)	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (2.5% POWER)
TYPES		
1)	1.E-4++	1.E-4++
1)'	1.E-3++	1.E-4++
2)	5.E-3	1.E-4++
3)	.25	.06
4)	.06	.02
5)	0.13	.03

\+ Based on containment conditions

++ Estimates of recovery probability at times greater than 24 hours are based upon engineering judgement since insufficient data exists to characterize such recovery probabilities even on a generic basis.

E.2.2.2 LOSS OF COOLANT ACCIDENTS (LOCAs)

At 2.5% power, the viability of CRD pump flow has its principal impact on mitigation of large and medium LOCAs sequences involving failures of low pressure systems. Using an estimated CRD injection reliability of 0.9/demand (similar to the small LOCA case in the Main Report), the estimated frequency of core vulnerable conditions is reduced to 9.4E-8 events/reactor year and 6.0E-8

events/reactor-year for large and medium LOCAs, respectively. Neither the additional time available nor the reduced challenge to plant systems from that described for 5% power is judged to have a quantifiable impact on other LOCA initiated core vulnerable sequences. Therefore, the total estimated frequency of core vulnerable conditions due to LOCAs initiated from 2.5% power is reduced to 2.2E-7 events/reactor-year.

E.2.2.3 OTHER TRANSIENTS

A significant contributor to the frequency of accidents initiated by "other" transients at 5% power is the failure to maintain primary system integrity. Transient induced LOCAs or SORVs tend to accelerate core heatup timing because coolant is lost early in the sequence. At 2.5% power, CRD flow is considered a viable alternative for coolant injection in these cases. Assuming credit for CRD flow similar to other accident sequences in this category at 5% power (i.e., a reliability of 0.99/demand), then the frequency estimates for induced LOCA or SORV accident sequences are reduced by two orders of magnitude.

Other dominant contributors to the frequency of core vulnerable conditions are postulated to involve failures of depressurization systems which prevents injection by low pressure systems. Intuitively, the extended core heatup timing provides a basis for arguing that failures of depressurization systems may be recovered prior to reaching unacceptably high fuel temperatures. However, the data and modeling required to support this assertion is a level of effort beyond the scope of this analysis. Therefore, it is judged that the already high combined reliability of depressurization systems at 5% power is adequate for the estimated reliability at 2.5% power. Thus, the total core vulnerable frequency due to other transients at 2.5% power is estimated to be 3.5E-7 events/reactor-year.

E.2.2.4 ATWS

The quantification of ATWS event trees at 5% power in the Main Report includes several changes in success criteria based on sequence timing. The differences between 5% and 2.5% power are judged to be negligible with regard to the

requantification of events appearing in ATWS sequences. Therefore, the estimated frequency of core vulnerable conditions is the same for both ones.

E.2.3 COMPARISON AND SUMMARY

Table 3-2 summarizes the quantification of accident sequences frequencies at 2.5% power. As shown, the total frequency of core vulnerable conditions is reduced by approximately a factor of 3, primarily due to the extended timing of LOSP sequences. An additional contributor is the assumed viability of CRD injection as a means of core cooling, which is also attributed to the extended sequence timing. While not explicitly calculated, it is judged that the extended sequence timing would have a significant favorable impact on other parameters important to risk calculations including: evacuation warning times, in-containment residence times, etc.

TABLE 3-2

DOMINANT ACCIDENT SEQUENCE FREQUENCIES ASSUMING STEADY STATE OPERATION AT 2.5° POWER

INITIATOR TYPE	TOTAL SEQUENCE FREQUENCY
Loss of Offsite Power	7.7E-7
LOCAs	2.2E-7
Other Transients	3.5E-7
ATWS	2.7E-7
TOTAL	1.6E-6

REFERENCES

1. Supplemental Motion for Low Power Operating License, submitted in the matter of Long Island Lighting Company, Docket No. 50-322, to the Atomic Safety and Licensing Board, affadavit of J. A. Notaro and W. E. Gunther, Jr., dated March 20, 1984.

2. Reactor Safety Study, WASH-1400, NUREG75/014, dated October 1975.

Transient induced LOCAs are significant in that they tend to be more dominant probabilistically than small or medium LOCAs. From a plant response perspective, these sequences are important since the rate of coolant lost through the safety relief valve could lead to rapid core uncovering. If coolant inventory is not reestablished, the core could heat up and core damage could occur early.

Following full power operation, the decay heat rate is sufficiently high that the reactor pressure could stabilize above the minimum HPCI and/or RCIC operational range, therefore, continued coolant injection by the HPCI and/or RCIC pumps is possible.

For low power operation, while the decay heat rate is low, the reactor operating pressure is nominally the same as for full power operation. Since steam discharge through the SORV is dependent upon the pressure, the rate of coolant loss for a given RPV pressure in both cases are equivalent.

A negative effect of the low decay heat rate for low power operation is that the reactor pressure may not be maintained above the minimum operating range of the steam driven HPCI and RCIC pumps given a SORV. Therefore, these pumps may become unavailable much sooner for LOSP SORV cases initiated at low power than for full power operation. The positive impact of the low decay heat rate, however, outweighs the negative effect in that a substantial reduction in the coolant makeup capacity is required to maintain sufficient core cooling.

In this analysis, several cases of Category 2 transient events were investigated:

(1) Stuck open relief valve (SORV) where the RPV coolant inventory lost through the SORV is not recovered

(2) SORV where coolant makeup through the high pressure injection systems is available initially.

(3) Transient events with immediate blowdown through the ADS valves, without coolant makeup. This case is judged to encompass large LOCA sequences in the steam lines.

(4) Transient events with controlled blowdown through the SRV, at a rate of less than 100°F/hour. The blowdown is initiated at 30 minutes in accordance to a procedural limit on drywell conditions.

For those sequences where inventory lost through the SORV is assumed to be replenished, the plant models considered that the coolant injection is initially provided by the HPCI and/or RCIC pumps during depressurization (while the reactor pressure is above the required set points). These sequences were further examined with minimal coolant make-up. It was assumed that RPV inventory was subsequently maintained by the CRD pumps in the long term following reactor depressurization and the single cycle of HPCI injection.

For this category of transient events, the reactor depressurization would leave the core uncovered. Without coolant makeup, the core would then heat up. Due to some steam cooling during the depressurization stage, radiative heat transfer and low decay heat levels, core vulnerability is estimated not to occur until after 2 to 7 hours following the transient event at five percent power operation. This would be extended further if the steam driven pumps are given credit and coolant injection occurs prior to reactor depressurization below the HPCI pump operational range.

Case 1: SORV Without Coolant Makeup

The particular accident sequence evaluated here assumed the reactor is shutdown and isolated following the initiating event. The RCS coolant temperature increases and the reactor pressure approaches the SRV setpoint of 1130 psia. The SRVs open to relieve reactor pressure and one SRV fails to close. Due to the low power levels of the core, the initially lower RPV pressure, and the initially lower core average temperature, the RCS coolant heatup to the SRV setpoint does not occur almost immediately as would be expected during the same transient event initiated at full power. In this analysis, it is estimated that the SRV setpoint would be reached

after approximately 20 minutes following the initiating event. The MARCH analysis predicts core uncovery occurring at 25 to 30 minutes, and core heatup is initiated with the core fully uncovered. Core overheat occurs in a steam starved environment. The reactor is depressurized and the coolant level is below the bottom active fuel height. These factors lead to minimal steam generation rates, thus cladding oxidation does not contribute significantly to core overheating. The estimated time to reach core vulnerability is approximately 4 hours following the initiating event for this case of Category 2 accident sequences.

Case 2: SORV With Coolant Makeup

This case of transient induced LOCA sequences involves a similar transient event as Case 1 described above. In Case 2, following the SORV, during the reactor depressurization, the HPCI pump is assumed to be activated automatically. Because of the coolant loss due to the combined effect of flashing and boil off, it is assumed that water level in the reactor does not exceed Level 8 thus the HPCI pumps are not tripped prematurely. It is further assumed that the HPCI pumps would continue to operate as long as the reactor pressure is above 150 psig. Therefore, the boundary conditions for coolant boil-off would be a reactor at 150 psig, the water level at approximately NWL, and a stuck open relief valve. Reactor depressurization continues until the reactor pressure drops to approximately 20 psia. Boil-off of the remaining coolant inventory at 20 psia involves a longer time than would occur at an elevated pressure due to the higher heat of vaporization at the lower pressure compared to 1130 psia.

In this scenario, following the initial core recovery, without subsequent coolant makeup from other modes of coolant injection, it was determined that coolant boil off to the top of active fuel would occur within 20-25 hours. Subsequent core vulnerability is not expected to occur until after 12 to 15 hours. Therefore, with a more realistic assessment of SORV sequences for LOSP cases, there is a potential for extending the time

before core vulnerability occurs. Since there exists a driving force for the HPCI/RCIC pumps (i.e., sufficient steam pressure and flow rate from the RPV during reactor depressurization), it is judged that there is sufficient time to recover the other modes of coolant injection. The CRD pumps providing subsequent coolant flow into the vessel was found to mitigate this accident event.

Case 3: Transient Events with Immediate Blowdown

Case 3 is the limiting scenario of the Category 2 accident events involving transient induced LOCA accident sequences. This scenario is intended to encompass large LOCA accident sequences as well, in which the reactor is depressurized to containment conditions almost immediately. As in the SORV cases, the core is completely uncovered after blowdown. But since blowdown occurs very rapidly, the stored thermal energy in the core is not sufficiently dissipated prior to initiation of core overheat. In this sequence, the core becomes vulnerable within 2-1/2 to 3 hours, following the initiating event.

Case 4: Transient Event with Controlled Blowdown

This scenario considers a controlled depressurization of the reactor initiated by the operator to meet procedural limits of drywell temperature. During loss of all AC power sequences, the drywell coolers become unavailable, and the drywell atmosphere could heat up beyond 296°F which would require reactor depressurization. The drywell temperature climbs rapidly during the first 5 to 10 minutes before heat transfer to the drywell liner is established. In this analysis, the drywell temperature limit is assumed to be reached within 30 minutes at which point the reactor is depressurized at a rate not to exceed 100°F per hour. This scenario effectively extends initiation of core heatup due to the concommitant steam cooling during the depressurization stage. Although core uncovery occurs within a few minutes, the blowdown to a pressure of

150 psig* is reached at approximately 150 minutes following the initiating event. Core heatup subsequently follows and core vulnerability could occur after approximately 6-7 hours.

B.3.3 Category 3 - Anticipated Transients Without Scram

This category of accident sequences include those low frequency event sequences in which an anticipated transient coupled with failure to insert the control rods may occur. In the Shoreham PRA, the evaluation of postulated ATWS accident events indicated that these sequences could potentially lead to a more severe containment challenge compared with the other types of accident sequences investigated thus far. Following operation at 100% power, pool heat-up and containment overpressure occurs rapidly. The estimated time of core vulnerability determined in the PRA was approximately 30-40 minutes.

For ATWS events for which the condenser is isolated, the initial low power level of 5 percent results in a slower rate of suppression pool heat up which then provides more time for the operator to take action in mitigating the accident. In this evaluation, it is estimated that several hours would be required to heat up the suppression pool to saturation and several more hours for the containment to reach its ultimate pressure capacity. This assumes that the reactor power drops to approximately three percent of rated power or 60 percent of the initial low power level of five percent. RCS coolant inventory makeup is derived from the turbine driven HPCI pumps, and both trains of the residual heat removal heat exchangers are postulated to be unavailable. The significance of high suppression pool temperature relates to the operability of the HPCI pumps under adverse conditions, e.g., lube oil

* The emergency procedure guidelines call for the operator maintaining the reactor pressure at 100-150 psig. It is assumed in this evaluation that the operator would tend to keep the reactor pressure from dropping well below 150 psig. This would allow sufficient driving force for the steam turbine HPCI pump driver thereby ensuring availability of the HPCI system.

degradation. For some ATWS sequences evaluated in the PRA, continued coolant injection may jeopardize containment integrity which in turn could result in the degradation of the ECC systems and inability to maintain coolant inventory. Therefore, two cases of ATWS isolation events were considered in this evaluation:

(1) Coolant injection is terminated at the point when the suppression pool temperature reaches 240°F. In this scenario, coolant boil off would occur in an intact containment.

(2) Continued coolant injection is assumed, leading to containment failure and ECC failure. This is subsequently followed by boil off of RCS coolant inventory in a failed containment.

In both cases described above, core vulnerability would follow shortly after core uncovery and dryout at decay heat energy levels.

The calculations performed predict an estimated time of approximately 3-4 hours to heat up the suppression pool to 240°F. For Case 1, subsequent to coolant injection being terminated at this point, the RPV inventory to the TAF is boiled off with the reactor at 3 percent of rated power. The core is uncovered within 30 minutes after termination of coolant injection and heats up at the fission product decay heat power. It is estimated that core vulnerability would occur at greater than 5-6 hours following core uncovery. For Case 2, assuming that coolant injection is maintained, the containment is estimated to fail by overpressure after approximately 6-7 hours. Following containment failure, ECCS injection is terminated. Coolant boil off is estimated to uncover the core after 30 minutes and core overheat would subsequently occur after approximately 5-6 hours. In this analysis, the energy released via the steam flow to the Terry turbine HPCI/RCIC pump driver was not considered. It is estimated that pool heatup and containment failure could be delayed by approximately 5-10 percent if this additional heat sink is considered in the analysis.

Transient events with failure to shutdown the reactor is potentially a more severe accident category because of the higher thermal energy release rate compared to shutdown conditions. This category of accident events result in

the SRVs being open more often than would be if the reactor were shutdown. The cycling of the SRVs present a challenge that could lead to one of the SRVs failing to close. An ATWS event coupled with a SORV was investigated to determine the impact of the stuck open relief valve on plant response during the ATWS event. It was determined that the reactor pressure could stabilize at approximately 350 to 400 psia for this sequence. The impact on sequence timing would not be noticeably different. However, a significant perturbation would be the reactor coolant loss following termination of injection flow. Blowdown could occur from 350 psi to containment conditions due to the stuck open relief valve following decay in the core power which could result in a more rapid core uncovering. If the core becomes fully uncovered, a steam starved core condition would be expected during the core heat up phase delaying the core heat up period to some extent due to reduced clad oxidation.

B.4 Sequence Perturbations

The three categories of accident sequences described above were also examined to determine the sensitivity of the accident event timing for power levels less than the reference value of five percent. In this sensitivity evaluation, the reactor was assumed to be operated at a constant level of 2.5 percent of rated power; not the more likely scenario involving impulse power fluctuations below the reference five percent level. The accident progression determinations were conducted consistent with the five percent power accident analysis.

Because of the decaying nature of fission product energy release rate, the time scales of the accident sequences following shutdown are not always inversely related to the initial power level, i.e., at 2.5 percent power, the time to core vulnerability is not exactly twice that of the 5 percent power level case. In general, the required time to boil off the same amount of water inventory or heat up of the fuel would vary depending upon the time from shutdown at which boil off or core heat up is initiated. This is apparent from the results of the assessment of the sequence perturbations of the

category 1 transient events (isolation without makeup) initiated at five percent power level described in Section B.3.1.

The results of this sensitivity evaluation indicate that for the transient events and isolation cases without coolant makeup, (event category 1 described above) the time to initial core uncovering estimated for the 2.5 percent power level case is approximately 45 to 50 hours, and the subsequent core vulnerability is estimated to occur after another 26 to 30 hours. If the reactor inventory is assumed recovered after boil off to Level 2, the time to core vulnerability would be found extended accordingly. This assumes only a single cycle of HPCI and that subsequent coolant injection does not occur. Event category 2 evaluation (transient induced LOCAs) shows the same trend, and core overheat is predicted to occur at greater than 10 hours.

For the ATWS scenarios, for which no containment heat removal capability is assumed, containment integrity is jeopardized and coolant injection is lost at 6-7 hours. This is followed by boil off and core overheat occurring after another 8 hours after containment failure. For this sequence involving failure to bring the reactor subcritical, the initial phase of the accident assumes that the reactor is at 60 percent of initial power level. Therefore, the rate of suppression pool heatup and containment pressurization is expected to be directly related to the assumed power level.

B.5 Summary and Conclusions

The accident analysis performed for the potentially risk dominant sequences described above indicates that a significant risk reduction (in terms of estimated frequencies) during low power operation is possible. The substantial time required to reach core vulnerability for each event category discussed in Section B.3, range from 2-3 hours to several days. This range of times would most likely provide sufficient time for operator action to mitigate the

accident. In addition, the required mitigative capacity of coolant injection sources is significantly reduced such that other coolant injection success paths are possible.

The results of this evaluation as summarized in Table 3 provides an indication of the time windows available for the operators to implement mitigative actions. This table also shows the range of times prior to core vulnerability for the accident sequences studied in this appendix given some perturbations in the time delay prior to initiation of coolant boil off. It appears that at low power levels, the time scales of the accident sequence progression to the point where the core or containment integrity may be lost are quite long that substantial times are available for operator action to mitigate the accident.

From an overall risk perspective (i.e., frequency and consequence considerations), the potential off-site public health impact of these accident sequences initiated at low power would be reduced significantly. Because of the low decay heat energy release rate, the containment integrity will likely be maintained for several days given that the accident does proceed unchecked to a core meltdown. This will undoubtedly remove significant portions of the airborne radionuclide materials from containment, thus substantially reducing the amounts of fission products that could be released to the environment. On the basis of this assessment, and considering the general aspects of radionuclide behavior, it is concluded that significant reduction in the source terms is possible at low power operation. It is estimated that for the extended times prior to containment failure and radionuclide release, a source term reduction factor ranging from 10 to 100 may be possible for similar accident sequences involving early containment failures studied in the PRA. Furthermore, the fission product inventory of the reactor core operated at low power restricted to five percent is a factor of 20 less than that at full power.

Table 3
SUMMARY OF EVENT TIMING FOR SELECTED ACCIDENT SEQUENCES AT LOW POWER OPERATION

ACCIDENT CATEGORY	TIME DELAY PRIOR TO BOIL OFF	TIME TO CORE UNCOVERY FROM BOIL OFF INITIATION	TIME TO CORE OVERHEAT FROM CORE UNCOVERY	TOTAL TIME TO CORE VULNERABILITY
CATEGORY 1				
Isolation	0	18 hours	11-13 hours	30 hours
Transients	4 hours	29 hours	12-14 hours	2 days
	10 hours	33 hours	13-15 hours	2-1/2 days
	30 hours	35 hours	14-18 hours	3-1/2 days
CATEGORY 2				
SORV w/o makeup	0	25-30 minutes	3-1/2 hours	4 hours
SORV w/initial makeup	30 minutes	20-25 hours	12-14 hours	1-1/2 days
ADS (LOCA)	0	0	2-3 hours	3 hours
Controlled Blowdown	0	45-60 minutes	6-7 hours	7.5 hours
CATEGORY 3				
ATWS with containment intact	3-4 hours	30 minutes	5-6 hours	8-10 hours
ATWS with containment failed	6-7 hours	30 minutes	5-6 hours	11-13* hours

* Time from initiating event given continued coolant injection from the suppression pool, i.e., the impact of high pool water temperature (greater than 240°F) on HPCI/RCIC operability is not considered in the analysis.

This analysis focused on LOSP cases for which the plant systems considered for coolant injection did not include those requiring off-site power. Extrapolating the results of this analysis to other accident initiators, could likewise lead to extended times available for operator action for similar accident sequence progression. In summary, a significant reduction of risk to the public (conservatively estimated as at least a factor of 20 to possibly 200 due to the reduced potential off-site consequences alone) is judged likely for the spectrum of accident events analyzed in the PRA for low power operation at Shoreham.

REFERENCES

1. Shoreham Nuclear Power Station, Probabilistic Risk Assessment, SAI-372-83-PA-01, June 1983

2. Draft Report on Status of Validation of the MARCH-2 Computer Code. Battelle Columbus Laboratories, Columbus, Ohio, July 11, 1983

3. Division of Systems and Reliability Research, Office of Nuclear Regulatory Research, Battelle Columbus Laboratories, MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual, R.O. Wooten, H.I. Avci, NUREG/CR-1711 BMI-2064, Columbus, Ohio, October 1980

4. LILCO Interoffice Memo, NFD-83-016, January 24, 1983

5. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, August 29, 1979

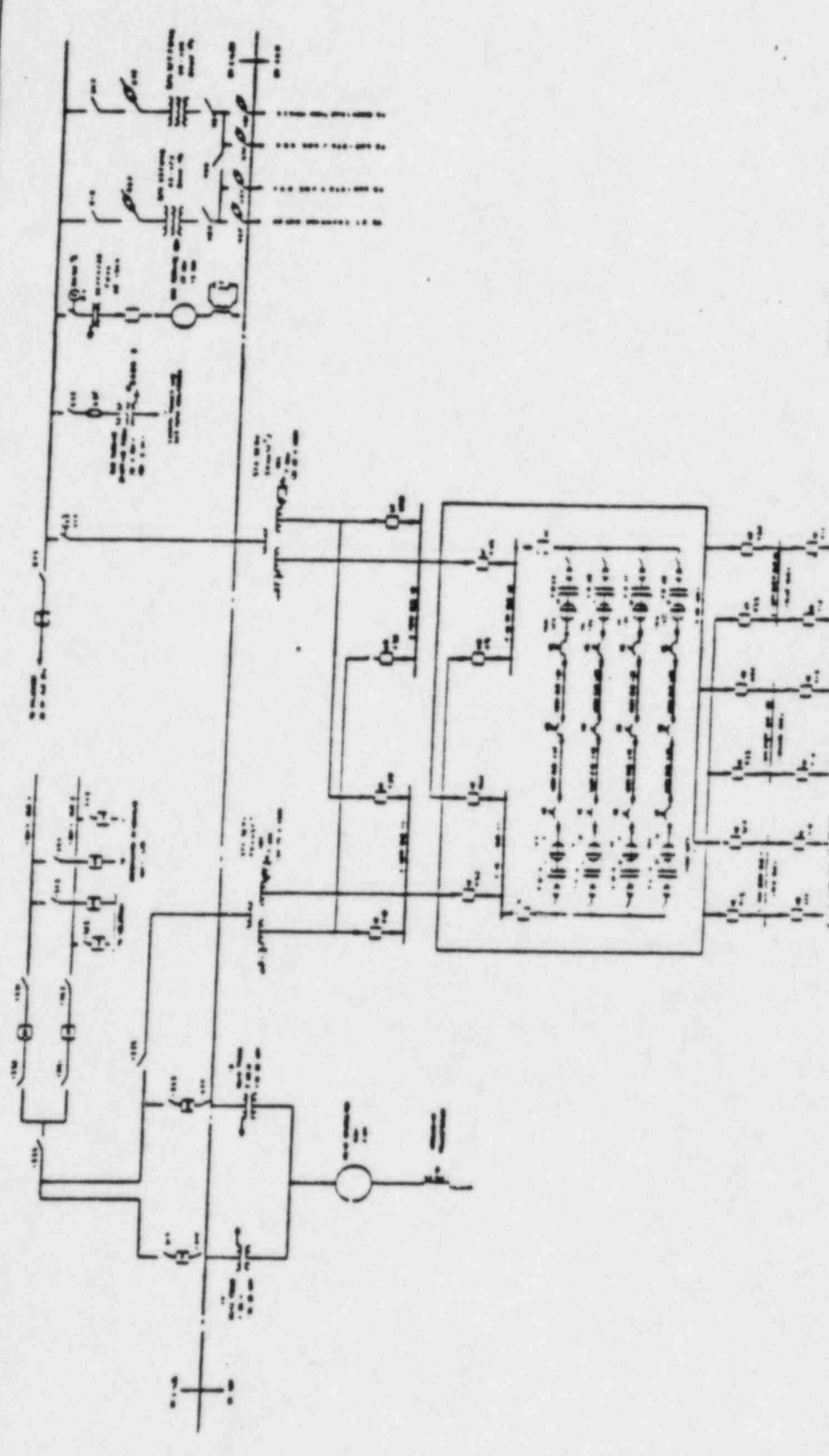
6. Personal Communication, R.J. Paccione (LILCO) and Z.T. Mendoza (SAI) dated March 22, 1984

APPENDIX C

SNPS/LILCO GRID ELECTRIC POWER SYSTEM DESCRIPTION

ere are two off-site power sources at SNPS which are physically and ectrically independent [1]. The primary source of off-site power to the ant is via the Normal Station Service (NSS) transformer which is connected tween the SNPS generator circuit breaker and the 138KV switchyard. The condary source is through the Reserve Station Service (RSS) transformer ich is connected to the 69KV transmission system. A schematic diagram of e 138KV transmission system in the area surrounding SNPS is shown in Figure 1. A one line diagram of the main portion of the SNPS electric power system shown in Figure C-2. The 138KV and 69KV transmission lines from the plant tend out to various substations at nearby locations on the LILCO grid. One the grid connections, the Holbrook Substation, currently has a connection a gas turbine generator with black start capability.

e on-site gas turbine generator considered for the black start modification SNPS is connected to the 69KV system on the site near the RSS transformer. e configuration of the 69KV system with respect to the on-site gas turbine shown in Figure C-3.



PORTIONS ARE NUCLEAR SAFETY RELATED

Figure C-2

MAIN ONE LINE DIAGRAM

SHOREHAM NUCLEAR POWER STATION-UNIT 1

REVISION 20 DECEMBER 1981

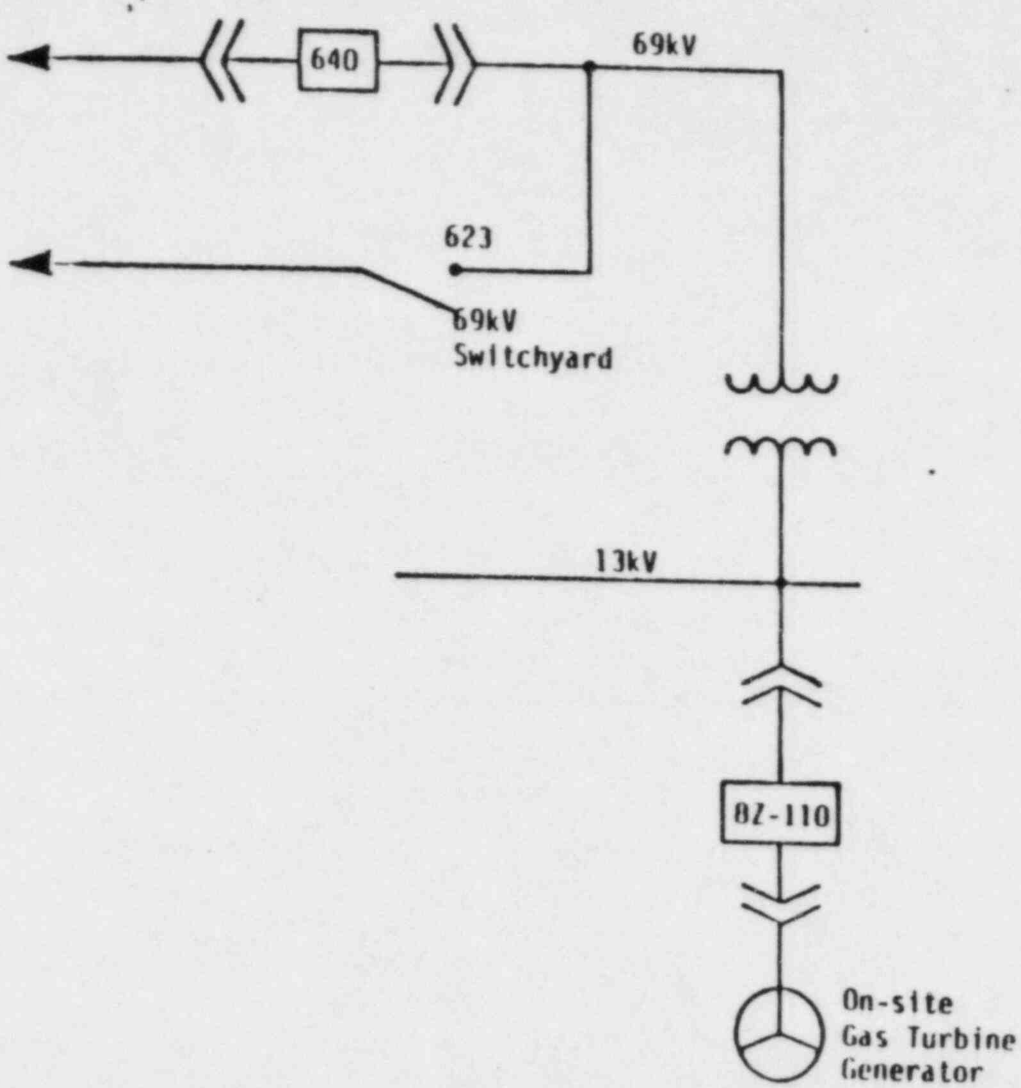
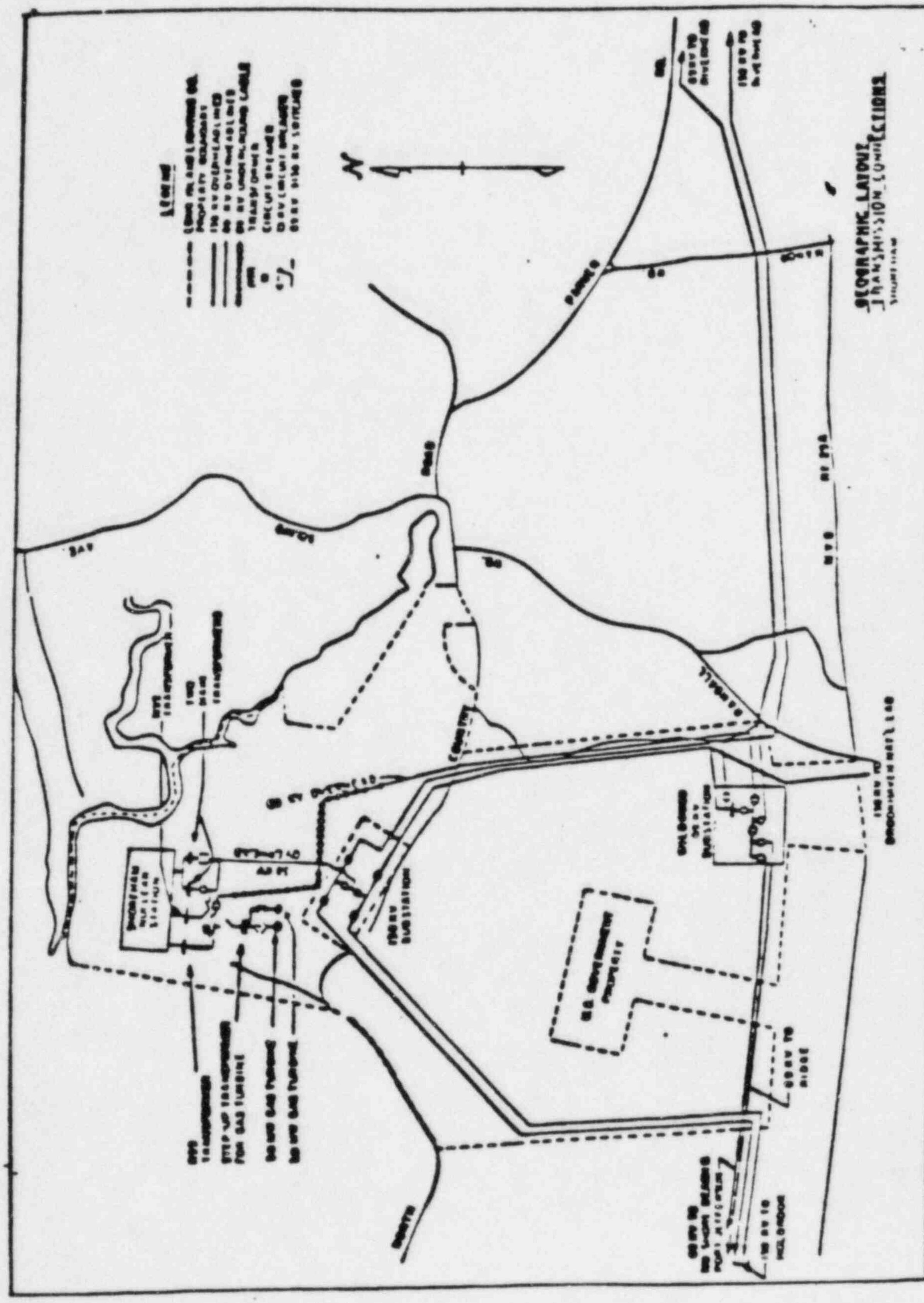


Figure C-3 Schematic Diagram of Shoreham 69KV Switchyard


GEOGRAPHIC LAYOUT
TRANSMISSION CONNECTIONS

REFERENCES

1. Shoreham Nuclear Power Station, Final Safety Analysis Report, Docket No. 50-322.

2. Letter from R.J. Paccione/R.S. Zambratto (LILCO) to E.T. Burns (WLA), April 9, 1983.

APPENDIX D

ASSESSMENT OF LOSP EVENT DATA AND APPLICATION OF DOMINANT ACCIDENT SEQUENCES

Two of the key parameters used in the Shoreham PRA for estimating the frequency of core vulnerable conditions following LOSP are: (1) the frequency of LOSP events, and (2) the time required for recovery of off-site power. These parameters are derived from the LILCO grid reliability data base and EPRI NP-2301, respectively. In the latter, data from a nationwide survey are examined and conditional probabilities of recovery events are estimated for Shoreham. The proposed black start modification provides a redundant and diverse method for restoration of off-site power.

The time required for recovery of off-site power depends heavily on the particular failure mode leading to the LOSP event. A review of the causes of the LOSP events in EPRI NP-2301 leads to the definition of three general categories of LOSP events: 1) grid failures or transmission line failures, 2) main switchyard failures, and 3) failures of both switchyards. These failure event categories allow the effectiveness of the black start gas turbine system at Shoreham to be modeled and a sensitivity of its effectiveness developed. The LOSP categories are described as follows:

<u>Grid or Transmission Line Failures:</u> Based upon a review of historical data throughout the U.S. [1,2], grid failures range from a total system shutdown to relatively minor switching errors that de-energize substations. With proper switching, either the black start system feeding the Holbrook Substation (22 miles from Shoreham) or the Shoreham on-site gas turbine with black start could restore power to Shoreham.

Transmission line failures are generally weather induced by causes such as wind storms or ice storms resulting in failures of several transmission lines. While the transmission lines at Shoreham are generally widely separated, the

69KV circuit does share the same right-of-way as one of the two 138KV circuits for a distance of approximately one mile. Therefore in cases in which the transmission lines become unavailable, the Holbrook black start system may be ineffective while the Shoreham on-site gas turbine could still provide power to the Shoreham on-site electrical distribution system.

In this evaluation it is judged that the likelihood of recovery from grid failures is similar for LILCO and for other grids in the U.S. In other words, the black start gas turbine capability at Holbrook is judged to be adequately included in the operating experience data base which reflects the strong possibility of recovery from such events. Therefore, in the assessment of the on-site black start gas turbine capability, both grid failures and transmission line failures can be lumped together for the purposes of quantification.

<u>Main Switchyard Failures:</u> These events apply to the range of failures which could occur in the vicinity of the main switchyard. These failures are considered to be independent of the availability of the 69KV system. In these cases, the 69KV system is expected to remain energized, so there is no substantial advantage to the addition of the on-site black start gas turbine.

<u>Failures of Both Switchyards:</u> These events involve common mode failures between the two switchyards. In these cases, the black start capability would not be effective since the 69KV switchyard is required to direct the power from the on-site gas turbine into the normal 4160V buses.

The allocation of the failure modes in EPRI NP-2301 into these three categories is crucial in the assessment of the relative worth of the on-site gas turbine and its potential public safety improvement. This allocation turns out to be one of the principal contributors to the uncertainty associated with the calculated reduction in core vulnerable frequency associated with the Shoreham design modification. Therefore, the following quantification is structured to provide a sensitivity on the best estimate values to indicate the potential variation based upon data uncertainty.

Table D-1 shows the categorization of the LOSP initiated events in EPRI NP-2301. Table D-1 has been constructed with some subjective judgement used to characterize the failure modes. In particular, the failure modes with potential impact on redundant switchyards have been inferred from the data; that is, those failure modes which are judged possible to cause a simultaneous failure of both Shoreham switchyards are identified. The conditional probability for failure to recover off-site power for each time phase is obtained from the sum of failures for each of the LOSP event categories. EPRI NP-2301 contains a large amount of useful data to provide an overview of what an "average" plant might look like. However, one must be prudent in the application of these data on a plant specific basis. In particular, associating the Shoreham-specific LOSP frequency with the generic failure modes from EPRI NP-2301 may underestimate the benefit of the black start capability.

Information presented in each column of Table D-1 is described below:

(1) Nuclear plant at which the LOSP is recorded.

(2) Incidents of LOSP which are caused by either total grid blackouts or transmission line failures [0 = less than 30 minute duration; x = is greater than 30 minute duration].

(3) Incidents of LOSP in which a main switchyard failure is involved.

(4) Incidents of LOSP recorded in column (3) which also involve multiple switchyard failures or the potential for such are identified in column (4). The probabilistic sensitivity analysis of Shoreham results in the calculation of two cases:

   A) Main switchyard failure considered to have a high likelihood of impacting the redundant switchyard.

   B) Main switchyard failure considered to have a high or uncertain likelihood of impacting the redundant switchyard.

(5) The duration of each LOSP event.

(6) The number of grid connections at each plant.

Table D-1

CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Beaver Valley		O			:17	2
Calvert Cliffs		X		X	5:29	2
Davis Besse		X*		X*	----	3
"		O			:26	3
Dresden 1	X				25:40	9
Farley		X		X	4:59	2
Fitzpatrick		O			<:01	2
"		O			:03	2
Fort Calhoun		X			11:05	7
"		X		X	:54	7
"		O			<:01	7
Ginna		O			:30	2
Ginna	X				:40	2
Haddam		O			:29	2
"		O			:09	2
"		O			<:01	2
"	O				:20	2
"		O			:16	2
Humbolt Bay		X*			----	2
Indian Point	X				6:28	2
La Crosse	O				:14	2
"		X			1:01	2
"		O			:20	2
"		O			:02	2
"		X		X	1:50	2
"		O			:10	2
"		X		X	5:35	2
Millstone 1		X	X	X	24:37	2
Nine Mile Point		O			<:01	2
Oconee		X			1:00	2
Oyster Creek		X*	X*	X*	----	3

See notes on following page.

Table D-1 (Continued)
CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Palisades		X			:56	5
"		X		X	4:45	5
"		X		X	3:30	5
"		X		X	1:30	5
Pilgrim	X				2:40	3
"	X				8:54	3
Point Beach		O			:08	2
"		X			:55	2
Quad Cities		X			1:11	4
San Onofre		X	X	X	4:59	7
"		O			:04	7
"		O			<:01	7
St. Lucie	O				:08	2
Yankee Rowe	X				:37	2

The total number of LOSP events is 45

X - Indicates an event lasting >30 minutes.
O - Indicates an event lasting <30 minutes.
* - Assumed, based on the specific failure mode.
Included in the calculations as a failure lasting >4 hours.

The recovery factors determined for LOSP events in this analysis incorporate some subtle but potentially important distinctions regarding the source of the data, i.e., the data set is composed of a spectrum of failures from rather minor single failure events through severe weather conditions affecting multiple transmission lines. Based upon the data in EPRI NP-2301 (causes and durations), it is clear that there is a strong coupling between the duration of the loss of off-site power outage and the particular failure mode. However, in the assessment of the on-site gas turbine capability the use of the data to allocate exact failure modes extends the statistical significance of the data to its limits. For this reason, this analysis attempts to examine the general trend of the coupling of the failure modes (e.g. weather) and

duration of LOSP and to provide both optimistic (CASE A) and pessismistic (CASE B) assumptions in interpretation of the data to account for uncertainties in classification of event. From this perspective, long duration transmission line failures take on an increasing importance. The analysis incorporates these data directly into the probabilistic evaluation including the strong coupling between failure mode and duration of LOSP.

Table D-2 summarizes the data and classifies the failure modes by time phase consistent with the PRA. The information in the table may be used to derive estimates for recovery values used in the recovery logic model.

The time phases which have the highest contribution to core vulnerable frequency following LOSP initiators, i.e., time phases III and IV, have a substantial fraction of events for which off-site power has not been recovered based upon historical nuclear plant operating experience.

Table D-2

SUMMARY OF LOSP EVENTS BY TIME PHA[illegible]

TIME PHASE	GRID AND OFF-SITE TRANSMISSION	MAIN SWITCHYARD	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS	
			CASE A (optimistic)	CASE B (pessimistic)
PHASE I > 30 MIN	6	19	3	12
PHASE II > 2 HR	4	11	3	9
PHASE III > 4 HR	3	10	3	8
PHASE IV > 10 HR	1	2	1	1

REFERENCES

1. Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301, March, 1982.

2. Scholl, R.F., "Loss of Off-Site Power Survey Status Repor", Revision 3, Report of the Systematic Evaluation of Program Brench, Division of Licensing, U.S. NRC.

APPENDIX E

SENSITIVITY STUDY OF THE CORE VULNERABLE FREQUENCY ASSOCIATED WITH LOW POWER TESTING

The Main Report provides an indication of the assumed plant configuration for low power testing. For simplification, steady state operation at 5% power is assumed. In reality, however, the plant will be in a state of flux as numerous system tests, inspections, and measurements take place. The assumption of a steady state power level of 5% is judged to be a conservative approach to the assessment of risk for low power operation. The purpose of this Appendix is to provide additional bases for concluding that this approach is conservative. Specifically, a summary discussion of the detailed operational aspects of low power testing [1] is provided along with a best estimate, rather than conservative upper bound, evaluation of the core vulnerable frequency.

E.1 DETAILED OPERATIONAL ASPECTS OF LOW POWER TESTING

The initial plant testing at Shoreham involves a series of four phases defined as follows:

<u>Phase I:</u> Fuel loading and precriticality testing
<u>Phase II:</u> Cold criticality testing
<u>Phase III:</u> Heatup and low power testing to rated pressure/temperature conditions (approximately 1% rated power)
<u>Phase IV:</u> Low power testing (1-5% rated power)

The structure of the testing program is such that the plant must fulfill specific testing and operational objectives in each phase before continuing to the next. Component "wear-in" failures will be repaired as they arise, which implies that the plant will have a full complement of safety systems available for each new power level.

In this evaluation, it is judged that essentially no measurable risk can be associated with the first two phases of operation. This conclusion is based on two observations:

JRH1

1. WASH-1400 [2] identified spent fuel handling accidents as a potential radionuclide release mechanism. Damage to unirradiated fuel would not fall into this category, however. Fresh fuel bundles can be handled in open air and would not be subject to melting or significant radionuclide release.

2. Cold criticality testing produces a negligible amount of heat (0.0001% to 0.001% of rated thermal power, or a maximum of 24kw). Under these conditions, there is essentially no need for RPV inventory makeup systems. If the RPV is inadvertantly drained, the core will become subcritical. Therefore, postulated precursors to core damage (e.g., LOCA or failure to scram) would have a negligible impact on core integrity.

Thus, Phases III and IV, in which the RPV is pressurized, steam is generated, and coolant makeup systems are required is judged to be the region of operation in which the first measurable risk due to radionuclide release appears. The testing program for these phases consists of repeated gradual heatups to a maximum of 5% full power followed by controlled cooldowns. Only a small fraction of the total operating period is spent near 5% power. Because of these power cycles, it is judged that decay heat levels, which are dependent upon the power history, can best be modeled using an intermediate valve between 0% and 5% and assuming steady state operation. For this reason, the core vulnerable frequency associated with operation at 2.5% power is estimated in the following section.

E.2 CORE VULNERABLE FREQUENCY DUE TO STEADY STATE OPERATION AT 2.5% POWER

As with the analysis presented in the Main Report, phenomenological calculations of postulated accident progressions are an essential starting point for assessing success criteria and quantifying event probabilities. The analysis presented in this section is intended to (1) review the results of MARCH calculations for accidents initiated for 2.5% power (2) identify and reassess events which are significantly different in timing or magnitude from the 5% power case; and (3) quantify these differences to estimate the core vulnerable frequency. This approach is judged to be the most effective method of

evaluating the sensitivity of the assumed power level. The results will be directly comparable to accident frequency estimates derived in the Main Report (5% power) and in the PRA (100% power).

E.2.1 ACCIDENT PROCESS CALCULATIONS

Appendix B presents the results of process calculations from MARCH for accidents postulated during low power operation. The emphasis is primarily on the evaluation of parameters assumed for steady state operation at 5% power. In particular, the differences between 5% and full power include an upwardly skewed axial power profile, an average core temperature of 640°F (representing a reduction of 400°F), and a reduced reactor pressure of 950 psia. Given these differences, the key modeling difference between the 5% and 2.5% power level calculations is the reduction in initial and decay heat power levels. With this in mind, Table 1 presents a summary of the estimated timing of postulated accident sequences in which the core becomes vulnerable to melting.

TABLE E-1

ACCIDENT SEQUENCE TIMING SUMMARY

SEQUENCE	TOTAL TIME TO CORE VULNERABILITY (HRS.)	
	5% POWER	2.5% POWER
Scram, Isolation, Failure of Coolant Injection	30	71 - 80
Scram, Large LOCA, Failure of Coolant Injection	3	10
ATWS, Containment Initially Intact	8 - 10	14 - 15

These timing estimates are judged to have a significant impact on the plant system success criteria beyond that already included in the Main Report. Three principal effects are as follows:

1. CRD pump flow is considered a viable alternative coolant injection source in accidents accelerated by a loss of coolant inventory. This applies to SORV cases, medium LOCAs, and large LOCAs, but not RPV ruptures.

2. RCIC alone is a viable alternative for coolant injection during an ATWS (at 5% power, a combination of RCIC and CRD flow is judged necessary).

3. The reactor power level following ATWS with RPT is estimated to be within the capacity of 1 RHR heat exchanger. This is based on the assumption that the power level will decrease by approximately 40% following RPT. Even if a single loop of RHR failed to match the ATWS power level, the challenge to containment is expected to be substantially extended. In these instances, a rationale similar to that described in the Main Report for dismissal of Class II challenges is judged applicable, i.e., the very long period of time available prior to containment failure represents a risk below that which can be credibly quantified. Therefore, such cases are judged to have a negligible frequency.

These success criteria are incorporated into the quantification of core vulnerable accident sequence frequencies in the remainder of this section. Additionally, revised event success criteria due to the timing estimates will be discussed as they arise.

E.2.2 QUANTIFICATION OF ACCIDENT SEQUENCE FREQUENCIES

This section corresponds to Section 3 of the Main Report. As such, a discussion and quantification of each of the four initiator types is presented; based on the revised success criteria for operation at 2.5% power.

E.2.1.1 LOSS OF OFFSITE POWER INITIATOR

The LOSP tree shown in Figure 3.1 of the Main Report consists of an initial subtree used to define groups of sequences with similar timing, followed by subsequent subtrees used for modeling time dependent events. For the reassessment at 2.5% power, the only event requantified is Event R: Recovery of Offsite Power. Table 3-1 presents a comparison of the estimated recovery probabilities for the 5% and 2.5% power cases. This requantification results in a reduced core vulnerable frequency estimate of 7.7E-7 events/reactor-year for LOSP initiators.

TABLE 3-1 CONDITIONAL PROBABILITY OF RECOVERY OF OFFSITE POWER AS A TIME DEPENDENT FUNCTION

ACCIDENT SEQUENCES	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (5% POWER)	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (2.5% POWER)
TYPES		
1)	1.E-4++	1.E-4++
1)'	1.E-3++	1.E-4++
2)	5.E-3	1.E-4++
3)	.25	.06
4)	.06	.02
5)	0.13	.03

\+ Based on containment conditions

++ Estimates of recovery probability at times greater than 24 hours are based upon engineering judgement since insufficient data exists to characterize such recovery probabilities even on a generic basis.

E.2.2.2 LOSS OF COOLANT ACCIDENTS (LOCAs)

At 2.5% power, the viability of CRD pump flow has its principal impact on mitigation of large and medium LOCAs sequences involving failures of low pressure systems. Using an estimated CRD injection reliability of 0.9/demand (similar to the small LOCA case in the Main Report), the estimated frequency of core vulnerable conditions is reduced to 9.4E-8 events/reactor year and 6.0E-8

events/reactor-year for large and medium LOCAs, respectively. Neither the additional time available nor the reduced challenge to plant systems from that described for 5% power is judged to have a quantifiable impact on other LOCA initiated core vulnerable sequences. Therefore, the total estimated frequency of core vulnerable conditions due to LOCAs initiated from 2.5% power is reduced to 2.2E-7 events/reactor-year.

E.2.2.3 OTHER TRANSIENTS

A significant contributor to the frequency of accidents initiated by "other" transients at 5% power is the failure to maintain primary system integrity. Transient induced LOCAs or SORVs tend to accelerate core heatup timing because coolant is lost early in the sequence. At 2.5% power, CRD flow is considered a viable alternative for coolant injection in these cases. Assuming credit for CRD flow similar to other accident sequences in this category at 5% power (i.e., a reliability of 0.99/demand), then the frequency estimates for induced LOCA or SORV accident sequences are reduced by two orders of magnitude.

Other dominant contributors to the frequency of core vulnerable conditions are postulated to involve failures of depressurization systems which prevents injection by low pressure systems. Intuitively, the extended core heatup timing provides a basis for arguing that failures of depressurization systems may be recovered prior to reaching unacceptably high fuel temperatures. However, the data and modeling required to support this assertion is a level of effort beyond the scope of this analysis. Therefore, it is judged that the already high combined reliability of depressurization systems at 5% power is adequate for the estimated reliability at 2.5% power. Thus, the total core vulnerable frequency due to other transients at 2.5% power is estimated to be 3.5E-7 events/reactor-year.

E.2.2.4 ATWS

The quantification of ATWS event trees at 5% power in the Main Report includes several changes in success criteria based on sequence timing. The differences between 5% and 2.5% power are judged to be negligible with regard to the

requantification of events appearing in ATWS sequences. Therefore, the estimated frequency of core vulnerable conditions is the same for both ones.

E.2.3 COMPARISON AND SUMMARY

Table 3-2 summarizes the quantification of accident sequences frequencies at 2.5% power. As shown, the total frequency of core vulnerable conditions is reduced by approximately a factor of 3, primarily due to the extended timing of LOSP sequences. An additional contributor is the assumed viability of CRD injection as a means of core cooling, which is also attributed to the extended sequence timing. While not explicitly calculated, it is judged that the extended sequence timing would have a significant favorable impact on other parameters important to risk calculations including: evacuation warning times, in-containment residence times, etc.

TABLE 3-2

DOMINANT ACCIDENT SEQUENCE FREQUENCIES ASSUMING STEADY STATE OPERATION AT 2.5° POWER

INITIATOR TYPE	TOTAL SEQUENCE FREQUENCY
Loss of Offsite Power	7.7E-7
LOCAs	2.2E-7
Other Transients	3.5E-7
ATWS	2.7E-7
TOTAL	1.6E-6

REFERENCES

1. Supplemental Motion for Low Power Operating License, submitted in the matter of Long Island Lighting Company, Docket No. 50-322, to the Atomic Safety and Licensing Board, affadavit of J. A. Notaro and W. E. Gunther, Jr., dated March 20, 1984.

2. Reactor Safety Study, WASH-1400, NUREG75/014, dated October 1975.

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Department of Nuclear Energy

May 7, 1984

Dr. A. Thadani
Reliability and Risk Assessment Branch
Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ashok:

Owing to unanticipated delays caused by the extra time devoted to the review of the "flood-initiator", the diversion of a key reviewer's time (K. Shiu) to other -higher priority- project (FIN A-3366), and the lack of response to our Q-1 questions from LILCO, the completion date for Task 1 must be redefined.

The participation of Dr. Shiu to the review of the Shoreham PRA is absolutely necessary for the successful completion of the program. Dr. Shiu is, however, a key contributor to the remaining tasks of the review of the GESSAR-II PRA (FIN A-3366) and to the project "Guidance and Probabilistic Analyses and Evaluations" (FIN A-3758), to both of which you have assigned high priority. Given the present priorities of the various programs, we propose to stop work on the review of the Shoreham PRA and resume it at a later date (on or about June 15, 1984) with a revised milestone for Task 1, September 30, 1984. No cost increase is associated with this milestone change.

If you have any questions, please do not hesitate to contact me.

Warm regards,

Ioannis A. Papazoglou, Group Leader
Risk Evaluation Group

IAP/dm
cc: W. Kato
R. Bari
R. Hall
R. Youngblood
A. Buslik (NRC)
E. Chow (NRC)

Ed - Ashok told BNL that they could delay FIN A-3758 & to submit schedule for SHOREHAM on this issue.

A/22

Transient induced LOCAs are significant in that they tend to be more dominant probabilistically than small or medium LOCAs. From a plant response perspective, these sequences are important since the rate of coolant lost through the safety relief valve could lead to rapid core uncovering. If coolant inventory is not reestablished, the core could heat up and core damage could occur early.

Following full power operation, the decay heat rate is sufficiently high that the reactor pressure could stabilize above the minimum HPCI and/or RCIC operational range, therefore, continued coolant injection by the HPCI and/or RCIC pumps is possible.

For low power operation, while the decay heat rate is low, the reactor operating pressure is nominally the same as for full power operation. Since steam discharge through the SORV is dependent upon the pressure, the rate of coolant loss for a given RPV pressure in both cases are equivalent.

A negative effect of the low decay heat rate for low power operation is that the reactor pressure may not be maintained above the minimum operating range of the steam driven HPCI and RCIC pumps given a SORV. Therefore, these pumps may become unavailable much sooner for LOSP SORV cases initiated at low power than for full power operation. The positive impact of the low decay heat rate, however, outweighs the negative effect in that a substantial reduction in the coolant makeup capacity is required to maintain sufficient core cooling.

In this analysis, several cases of Category 2 transient events were investigated:

(1) Stuck open relief valve (SORV) where the RPV coolant inventory lost through the SORV is not recovered

(2) SORV where coolant makeup through the high pressure injection systems is available initially.

(3) Transient events with immediate blowdown through the ADS valves, without coolant makeup. This case is judged to encompass large LOCA sequences in the steam lines.

(4) Transient events with controlled blowdown through the SRV, at a rate of less than 100°F/hour. The blowdown is initiated at 30 minutes in accordance to a procedural limit on drywell conditions.

For those sequences where inventory lost through the SORV is assumed to be replenished, the plant models considered that the coolant injection is initially provided by the HPCI and/or RCIC pumps during depressurization (while the reactor pressure is above the required set points). These sequences were further examined with minimal coolant make-up. It was assumed that RPV inventory was subsequently maintained by the CRD pumps in the long term following reactor depressurization and the single cycle of HPCI injection.

For this category of transient events, the reactor depressurization would leave the core uncovered. Without coolant makeup, the core would then heat up. Due to some steam cooling during the depressurization stage, radiative heat transfer and low decay heat levels, core vulnerability is estimated not to occur until after 2 to 7 hours following the transient event at five percent power operation. This would be extended further if the steam driven pumps are given credit and coolant injection occurs prior to reactor depressurization below the HPCI pump operational range.

Case 1: SORV Without Coolant Makeup

The particular accident sequence evaluated here assumed the reactor is shutdown and isolated following the initiating event. The RCS coolant temperature increases and the reactor pressure approaches the SRV setpoint of 1130 psia. The SRVs open to relieve reactor pressure and one SRV fails to close. Due to the low power levels of the core, the initially lower RPV pressure, and the initially lower core average temperature, the RCS coolant heatup to the SRV setpoint does not occur almost immediately as would be expected during the same transient event initiated at full power. In this analysis, it is estimated that the SRV setpoint would be reached

after approximately 20 minutes following the initiating event. The MARCH analysis predicts core uncovery occurring at 25 to 30 minutes, and core heatup is initiated with the core fully uncovered. Core overheat occurs in a steam starved environment. The reactor is depressurized and the coolant level is below the bottom active fuel height. These factors lead to minimal steam generation rates, thus cladding oxidation does not contribute significantly to core overheating. The estimated time to reach core vulnerability is approximately 4 hours following the initiating event for this case of Category 2 accident sequences.

Case 2: SORV With Coolant Makeup

This case of transient induced LOCA sequences involves a similar transient event as Case 1 described above. In Case 2, following the SORV, during the reactor depressurization, the HPCI pump is assumed to be activated automatically. Because of the coolant loss due to the combined effect of flashing and boil off, it is assumed that water level in the reactor does not exceed Level 8 thus the HPCI pumps are not tripped prematurely. It is further assumed that the HPCI pumps would continue to operate as long as the reactor pressure is above 150 psig. Therefore, the boundary conditions for coolant boil-off would be a reactor at 150 psig, the water level at approximately NWL, and a stuck open relief valve. Reactor depressurization continues until the reactor pressure drops to approximately 20 psia. Boil-off of the remaining coolant inventory at 20 psia involves a longer time than would occur at an elevated pressure due to the higher heat of vaporization at the lower pressure compared to 1130 psia.

In this scenario, following the initial core recovery, without subsequent coolant makeup from other modes of coolant injection, it was determined that coolant boil off to the top of active fuel would occur within 20-25 hours. Subsequent core vulnerability is not expected to occur until after 12 to 15 hours. Therefore, with a more realistic assessment of SORV sequences for LOSP cases, there is a potential for extending the time

before core vulnerability occurs. Since there exists a driving force for the HPCI/RCIC pumps (i.e., sufficient steam pressure and flow rate from the RPV during reactor depressurization), it is judged that there is sufficient time to recover the other modes of coolant injection. The CRD pumps providing subsequent coolant flow into the vessel was found to mitigate this accident event.

Case 3: Transient Events with Immediate Blowdown

Case 3 is the limiting scenario of the Category 2 accident events involving transient induced LOCA accident sequences. This scenario is intended to encompass large LOCA accident sequences as well, in which the reactor is depressurized to containment conditions almost immediately. As in the SORV cases, the core is completely uncovered after blowdown. But since blowdown occurs very rapidly, the stored thermal energy in the core is not sufficiently dissipated prior to initiation of core overheat. In this sequence, the core becomes vulnerable within 2-1/2 to 3 hours, following the initiating event.

Case 4: Transient Event with Controlled Blowdown

This scenario considers a controlled depressurization of the reactor initiated by the operator to meet procedural limits of drywell temperature. During loss of all AC power sequences, the drywell coolers become unavailable, and the drywell atmosphere could heat up beyond 296°F which would require reactor depressurization. The drywell temperature climbs rapidly during the first 5 to 10 minutes before heat transfer to the drywell liner is established. In this analysis, the drywell temperature limit is assumed to be reached within 30 minutes at which point the reactor is depressurized at a rate not to exceed 100°F per hour. This scenario effectively extends initiation of core heatup due to the concommitant steam cooling during the depressurization stage. Although core uncovery occurs within a few minutes, the blowdown to a pressure of

150 psig* is reached at approximately 150 minutes following the initiating event. Core heatup subsequently follows and core vulnerability could occur after approximately 6-7 hours.

B.3.3 Category 3 - Anticipated Transients Without Scram

This category of accident sequences include those low frequency event sequences in which an anticipated transient coupled with failure to insert the control rods may occur. In the Shoreham PRA, the evaluation of postulated ATWS accident events indicated that these sequences could potentially lead to a more severe containment challenge compared with the other types of accident sequences investigated thus far. Following operation at 100% power, pool heat-up and containment overpressure occurs rapidly. The estimated time of core vulnerability determined in the PRA was approximately 30-40 minutes.

For ATWS events for which the condenser is isolated, the initial low power level of 5 percent results in a slower rate of suppression pool heat up which then provides more time for the operator to take action in mitigating the accident. In this evaluation, it is estimated that several hours would be required to heat up the suppression pool to saturation and several more hours for the containment to reach its ultimate pressure capacity. This assumes that the reactor power drops to approximately three percent of rated power or 60 percent of the initial low power level of five percent. RCS coolant inventory makeup is derived from the turbine driven HPCI pumps, and both trains of the residual heat removal heat exchangers are postulated to be unavailable. The significance of high suppression pool temperature relates to the operability of the HPCI pumps under adverse conditions, e.g., lube oil

* The emergency procedure guidelines call for the operator maintaining the reactor pressure at 100-150 psig. It is assumed in this evaluation that the operator would tend to keep the reactor pressure from dropping well below 150 psig. This would allow sufficient driving force for the steam turbine HPCI pump driver thereby ensuring availability of the HPCI system.

degradation. For some ATWS sequences evaluated in the PRA, continued coolant injection may jeopardize containment integrity which in turn could result in the degradation of the ECC systems and inability to maintain coolant inventory. Therefore, two cases of ATWS isolation events were considered in this evaluation:

(1) Coolant injection is terminated at the point when the suppression pool temperature reaches 240°F. In this scenario, coolant boil off would occur in an intact containment.

(2) Continued coolant injection is assumed, leading to containment failure and ECC failure. This is subsequently followed by boil off of RCS coolant inventory in a failed containment.

In both cases described above, core vulnerability would follow shortly after core uncovery and dryout at decay heat energy levels.

The calculations performed predict an estimated time of approximately 3-4 hours to heat up the suppression pool to 240°F. For Case 1, subsequent to coolant injection being terminated at this point, the RPV inventory to the TAF is boiled off with the reactor at 3 percent of rated power. The core is uncovered within 30 minutes after termination of coolant injection and heats up at the fission product decay heat power. It is estimated that core vulnerability would occur at greater than 5-6 hours following core uncovery. For Case 2, assuming that coolant injection is maintained, the containment is estimated to fail by overpressure after approximately 6-7 hours. Following containment failure, ECCS injection is terminated. Coolant boil off is estimated to uncover the core after 30 minutes and core overheat would subsequently occur after approximately 5-6 hours. In this analysis, the energy released via the steam flow to the Terry turbine HPCI/RCIC pump driver was not considered. It is estimated that pool heatup and containment failure could be delayed by approximately 5-10 percent if this additional heat sink is considered in the analysis.

Transient events with failure to shutdown the reactor is potentially a more severe accident category because of the higher thermal energy release rate compared to shutdown conditions. This category of accident events result in

the SRVs being open more often than would be if the reactor were shutdown. The cycling of the SRVs present a challenge that could lead to one of the SRVs failing to close. An ATWS event coupled with a SORV was investigated to determine the impact of the stuck open relief valve on plant response during the ATWS event. It was determined that the reactor pressure could stabilize at approximately 350 to 400 psia for this sequence. The impact on sequence timing would not be noticeably different. However, a significant perturbation would be the reactor coolant loss following termination of injection flow. Blowdown could occur from 350 psi to containment conditions due to the stuck open relief valve following decay in the core power which could result in a more rapid core uncovering. If the core becomes fully uncovered, a steam starved core condition would be expected during the core heat up phase delaying the core heat up period to some extent due to reduced clad oxidation.

B.4 Sequence Perturbations

The three categories of accident sequences described above were also examined to determine the sensitivity of the accident event timing for power levels less than the reference value of five percent. In this sensitivity evaluation, the reactor was assumed to be operated at a constant level of 2.5 percent of rated power; not the more likely scenario involving impulse power fluctuations below the reference five percent level. The accident progression determinations were conducted consistent with the five percent power accident analysis.

Because of the decaying nature of fission product energy release rate, the time scales of the accident sequences following shutdown are not always inversely related to the initial power level, i.e., at 2.5 percent power, the time to core vulnerability is not exactly twice that of the 5 percent power level case. In general, the required time to boil off the same amount of water inventory or heat up of the fuel would vary depending upon the time from shutdown at which boil off or core heat up is initiated. This is apparent from the results of the assessment of the sequence perturbations of the

category 1 transient events (isolation without makeup) initiated at five percent power level described in Section B.3.1.

The results of this sensitivity evaluation indicate that for the transient events and isolation cases without coolant makeup, (event category 1 described above) the time to initial core uncovering estimated for the 2.5 percent power level case is approximately 45 to 50 hours, and the subsequent core vulnerability is estimated to occur after another 26 to 30 hours. If the reactor inventory is assumed recovered after boil off to Level 2, the time to core vulnerability would be found extended accordingly. This assumes only a single cycle of HPCI and that subsequent coolant injection does not occur. Event category 2 evaluation (transient induced LOCAs) shows the same trend, and core overheat is predicted to occur at greater than 10 hours.

For the ATWS scenarios, for which no containment heat removal capability is assumed, containment integrity is jeopardized and coolant injection is lost at 6-7 hours. This is followed by boil off and core overheat occurring after another 8 hours after containment failure. For this sequence involving failure to bring the reactor subcritical, the initial phase of the accident assumes that the reactor is at 60 percent of initial power level. Therefore, the rate of suppression pool heatup and containment pressurization is expected to be directly related to the assumed power level.

B.5 Summary and Conclusions

The accident analysis performed for the potentially risk dominant sequences described above indicates that a significant risk reduction (in terms of estimated frequencies) during low power operation is possible. The substantial time required to reach core vulnerability for each event category discussed in Section B.3, range from 2-3 hours to several days. This range of times would most likely provide sufficient time for operator action to mitigate the

accident. In addition, the required mitigative capacity of coolant injection sources is significantly reduced such that other coolant injection success paths are possible.

The results of this evaluation as summarized in Table 3 provides an indication of the time windows available for the operators to implement mitigative actions. This table also shows the range of times prior to core vulnerability for the accident sequences studied in this appendix given some perturbations in the time delay prior to initiation of coolant boil off. It appears that at low power levels, the time scales of the accident sequence progression to the point where the core or containment integrity may be lost are quite long that substantial times are available for operator action to mitigate the accident.

From an overall risk perspective (i.e., frequency and consequence considerations), the potential off-site public health impact of these accident sequences initiated at low power would be reduced significantly. Because of the low decay heat energy release rate, the containment integrity will likely be maintained for several days given that the accident does proceed unchecked to a core meltdown. This will undoubtedly remove significant portions of the airborne radionuclide materials from containment, thus substantially reducing the amounts of fission products that could be released to the environment. On the basis of this assessment, and considering the general aspects of radionuclide behavior, it is concluded that significant reduction in the source terms is possible at low power operation. It is estimated that for the extended times prior to containment failure and radionuclide release, a source term reduction factor ranging from 10 to 100 may be possible for similar accident sequences involving early containment failures studied in the PRA. Furthermore, the fission product inventory of the reactor core operated at low power restricted to five percent is a factor of 20 less than that at full power.

Table 3
SUMMARY OF EVENT TIMING FOR SELECTED ACCIDENT SEQUENCES
AT LOW POWER OPERATION

ACCIDENT CATEGORY	TIME DELAY PRIOR TO BOIL OFF	TIME TO CORE UNCOVERY FROM BOIL OFF INITIATION	TIME TO CORE OVERHEAT FROM CORE UNCOVERY	TOTAL TIME TO CORE VULNERABILITY
CATEGORY 1				
Isolation	0	18 hours	11-13 hours	30 hours
Transients	4 hours	29 hours	12-14 hours	2 days
	10 hours	33 hours	13-15 hours	2-1/2 days
	30 hours	35 hours	14-18 hours	3-1/2 days
CATEGORY 2				
SORV w/o makeup	0	25-30 minutes	3-1/2 hours	4 hours
SORV w/initial makeup	30 minutes	20-25 hours	12-14 hours	1-1/2 days
ADS (LOCA)	0	0	2-3 hours	3 hours
Controlled Blowdown	0	45-50 minutes	6-7 hours	7.5 hours
CATEGORY 3				
ATWS with containment intact	3-4 hours	30 minutes	5-6 hours	8-10 hours
ATWS with containment failed	6-7 hours	30 minutes	5-6 hours	11-13* hours

* Time from initiating event given continued coolant injection from the suppression pool, i.e., the impact of high pool water temperature (greater than 240°F) on HPCI/RCIC operability is not considered in the analysis.

This analysis focused on LOSP cases for which the plant systems considered for coolant injection did not include those requiring off-site power. Extrapolating the results of this analysis to other accident initiators, could likewise lead to extended times available for operator action for similar accident sequence progression. In summary, a significant reduction of risk to the public (conservatively estimated as at least a factor of 20 to possibly 200 due to the reduced potential off-site consequences alone) is judged likely for the spectrum of accident events analyzed in the PRA for low power operation at Shoreham.

REFERENCES

1. Shoreham Nuclear Power Station, Probabilistic Risk Assessment, SAI-372-83-PA-01, June 1983

2. Draft Report on Status of Validation of the MARCH-2 Computer Code. Battelle Columbus Laboratories, Columbus, Ohio, July 11, 1983

3. Division of Systems and Reliability Research, Office of Nuclear Regulatory Research, Battelle Columbus Laboratories, MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual, R.O. Wooten, H.I. Avci, NUREG/CR-1711 BMI-2064, Columbus, Ohio, October 1980

4. LILCO Interoffice Memo, NFD-83-016, January 24, 1983

5. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, August 29, 1979

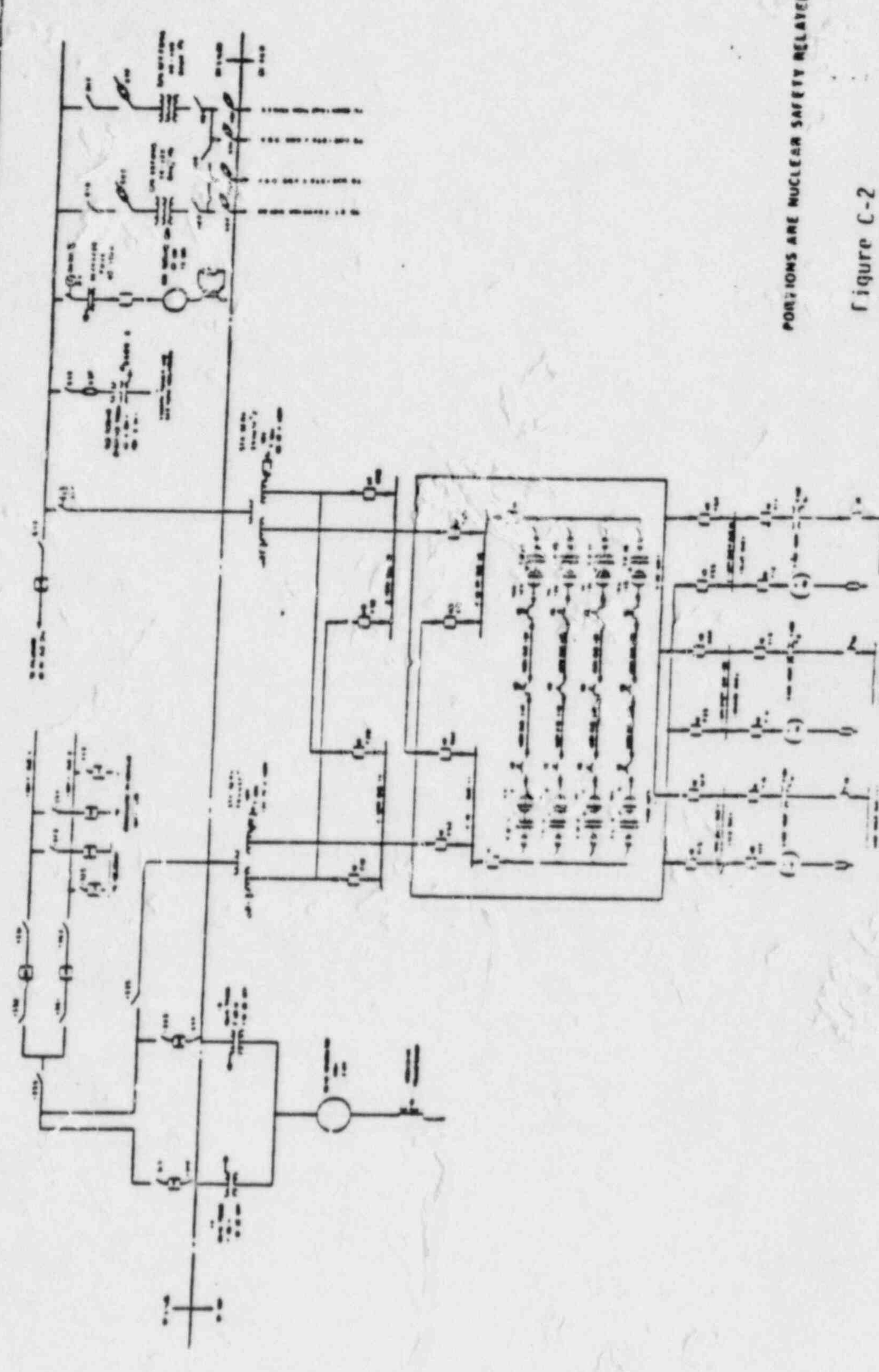
6. Personal Communication, R.J. Paccione (LILCO) and Z.T. Mendoza (SAI) dated March 22, 1984

APPENDIX C

SNPS/LILCO GRID ELECTRIC POWER SYSTEM DESCRIPTION

ere are two off-site power sources at SNPS which are physically and ectrically independent [1]. The primary source of off-site power to the ant is via the Normal Station Service (NSS) transformer which is connected tween the SNPS generator circuit breaker and the 138KV switchyard. The condary source is through the Reserve Station Service (RSS) transformer ich is connected to the 69KV transmission system. A schematic diagram of e 138KV transmission system in the area surrounding SNPS is shown in Figure 1. A one line diagram of the main portion of the SNPS electric power system shown in Figure C-2. The 138KV and 69KV transmission lines from the plant tend out to various substations at nearby locations on the LILCO grid. One the grid connections, the Holbrook Substation, currently has a connection a gas turbine generator with black start capability.

e on-site gas turbine generator considered for the black start modification SNPS is connected to the 69KV system on the site near the RSS transformer. e configuration of the 69KV system with respect to the on-site gas turbine shown in Figure C-3.



PORTIONS ARE NUCLEAR SAFETY RELATED

Figure C-2
MAIN ONE LINE DIAGRAM
SHOREHAM NUCLEAR POWER STATION-UNIT 1

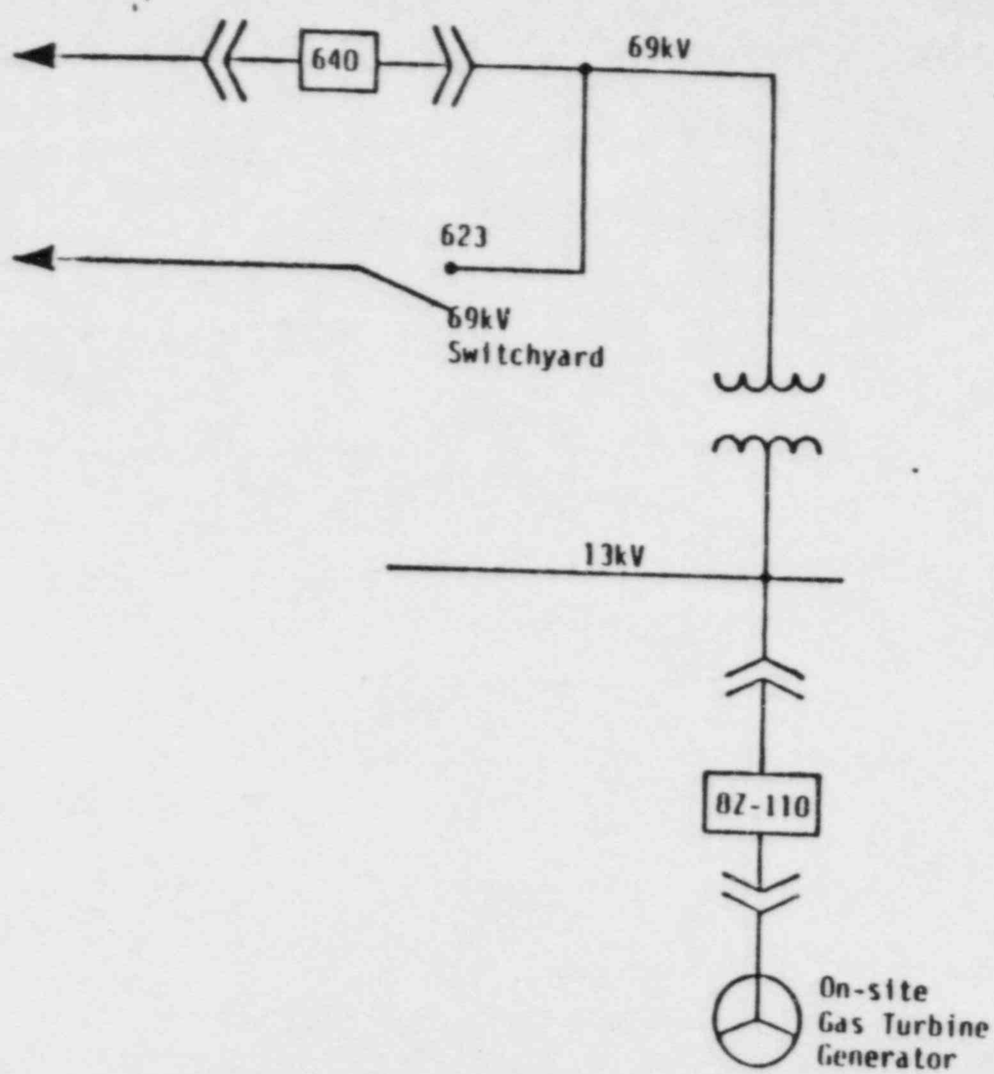
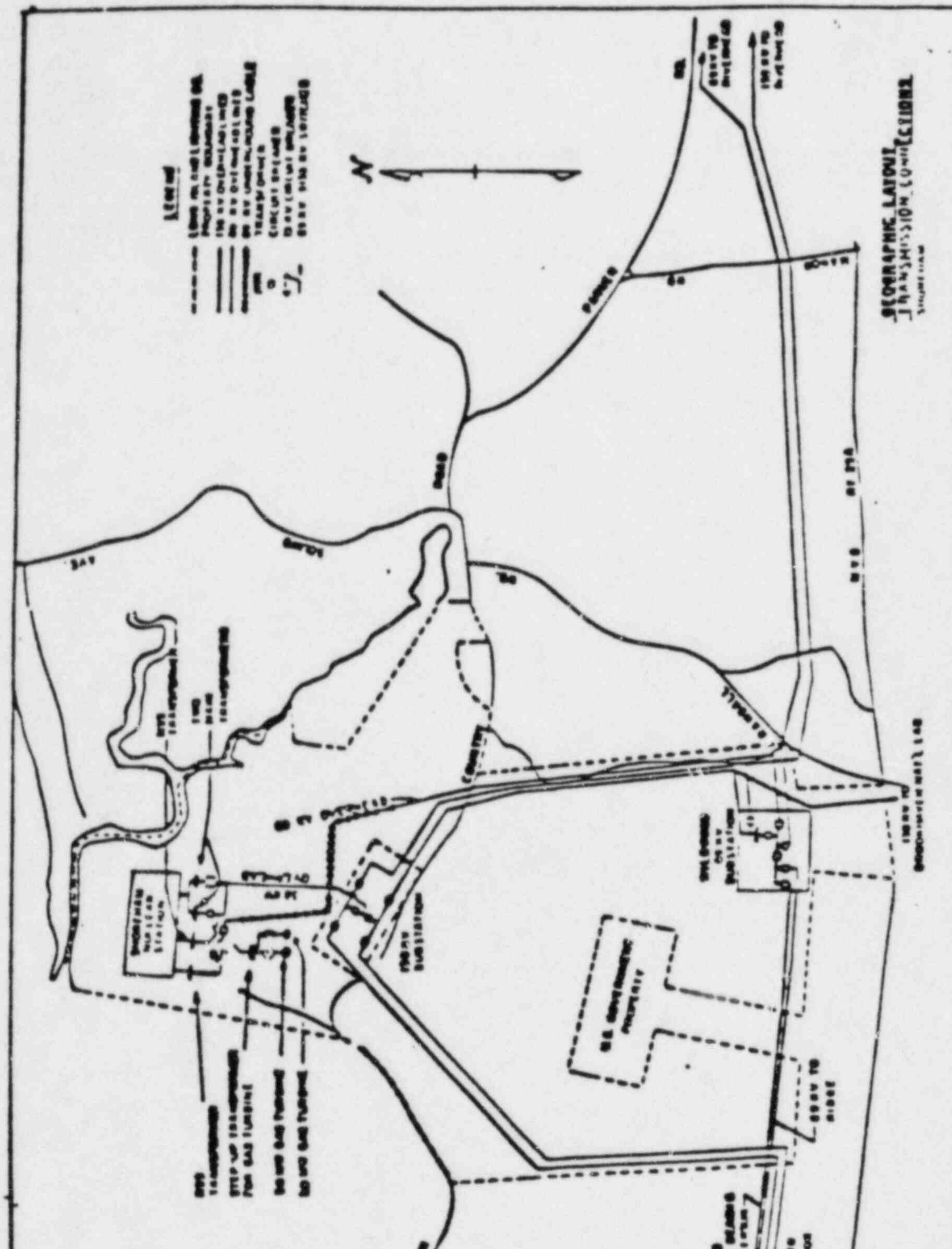


Figure C-3 Schematic Diagram of Shoreham 69KV Switchyard


GEOGRAPHIC LAYOUT
TRANSMISSION CONNECTIONS

REFERENCES

1. Shoreham Nuclear Power Station, Final Safety Analysis Report, Docket No. 50-322.

2. Letter from R.J. Paccione/R.S. Zambratto (LILCO) to E.T. Burns (WLA), April 9, 1983.

APPENDIX D

ASSESSMENT OF LOSP EVENT DATA AND APPLICATION OF DOMINANT ACCIDENT SEQUENCES

Two of the key parameters used in the Shoreham PRA for estimating the frequency of core vulnerable conditions following LOSP are: (1) the frequency of LOSP events, and (2) the time required for recovery of off-site power. These parameters are derived from the LILCO grid reliability data base and EPRI NP-2301, respectively. In the latter, data from a nationwide survey are examined and conditional probabilities of recovery events are estimated for Shoreham. The proposed black start modification provides a redundant and diverse method for restoration of off-site power.

The time required for recovery of off-site power depends heavily on the particular failure mode leading to the LOSP event. A review of the causes of the LOSP events in EPRI NP-2301 leads to the definition of three general categories of LOSP events: 1) grid failures or transmission line failures, 2) main switchyard failures, and 3) failures of both switchyards. These failure event categories allow the effectiveness of the black start gas turbine system at Shoreham to be modeled and a sensitivity of its effectiveness developed. The LOSP categories are described as follows:

Grid or Transmission Line Failures: Based upon a review of historical data throughout the U.S. [1,2], grid failures range from a total system shutdown to relatively minor switching errors that de-energize substations. With proper switching, either the black start system feeding the Holbrook Substation (22 miles from Shoreham) or the Shoreham on-site gas turbine with black start could restore power to Shoreham.

Transmission line failures are generally weather induced by causes such as wind storms or ice storms resulting in failures of several transmission lines. While the transmission lines at Shoreham are generally widely separated, the

69KV circuit does share the same right-of-way as one of the two 138KV circuits for a distance of approximately one mile. Therefore in cases in which the transmission lines become unavailable, the Holbrook black start system may be ineffective while the Shoreham on-site gas turbine could still provide power to the Shoreham on-site electrical distribution system.

In this evaluation it is judged that the likelihood of recovery from grid failures is similar for LILCO and for other grids in the U.S. In other words, the black start gas turbine capability at Holbrook is judged to be adequately included in the operating experience data base which reflects the strong possibility of recovery from such events. Therefore, in the assessment of the on-site black start gas turbine capability, both grid failures and transmission line failures can be lumped together for the purposes of quantification.

Main Switchyard Failures: These events apply to the range of failures which could occur in the vicinity of the main switchyard. These failures are considered to be independent of the availability of the 69KV system. In these cases, the 69KV system is expected to remain energized, so there is no substantial advantage to the addition of the on-site black start gas turbine.

Failures of Both Switchyards: These events involve common mode failures between the two switchyards. In these cases, the black start capability would not be effective since the 69KV switchyard is required to direct the power from the on-site gas turbine into the normal 4160V buses.

The allocation of the failure modes in EPRI NP-2301 into these three categories is crucial in the assessment of the relative worth of the on-site gas turbine and its potential public safety improvement. This allocation turns out to be one of the principal contributors to the uncertainty associated with the calculated reduction in core vulnerable frequency associated with the Shoreham design modification. Therefore, the following quantification is structured to provide a sensitivity on the best estimate values to indicate the potential variation based upon data uncertainty.

Table D-1 shows the categorization of the LOSP initiated events in EPRI NP-2301. Table D-1 has been constructed with some subjective judgement used to characterize the failure modes. In particular, the failure modes with potential impact on redundant switchyards have been inferred from the data; that is, those failure modes which are judged possible to cause a simultaneous failure of both Shoreham switchyards are identified. The conditional probability for failure to recover off-site power for each time phase is obtained from the sum of failures for each of the LOSP event categories. EPRI NP-2301 contains a large amount of useful data to provide an overview of what an "average" plant might look like. However, one must be prudent in the application of these data on a plant specific basis. In particular, associating the Shoreham-specific LOSP frequency with the generic failure modes from EPRI NP-2301 may underestimate the benefit of the black start capability.

Information presented in each column of Table D-1 is described below:

(1) Nuclear plant at which the LOSP is recorded.

(2) Incidents of LOSP which are caused by either total grid blackouts or transmission line failures [0 = less than 30 minute duration; x = is greater than 30 minute duration].

(3) Incidents of LOSP in which a main switchyard failure is involved.

(4) Incidents of LOSP recorded in column (3) which also involve multiple switchyard failures or the potential for such are identified in column (4). The probabilistic sensitivity analysis of Shoreham results in the calculation of two cases:

A) Main switchyard failure considered to have a high likelihood of impacting the redundant switchyard.

B) Main switchyard failure considered to have a high or uncertain likelihood of impacting the redundant switchyard.

(5) The duration of each LOSP event.

(6) The number of grid connections at each plant.

Table D-1

CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Beaver Valley		O			:17	2
Calvert Cliffs		X		X	5:29	2
Davis Besse		X*		X*	----	3
"		O			:26	3
Dresden 1	X				25:40	9
Farley		X		X	4:59	2
Fitzpatrick		O			<:01	2
"		O			:03	2
Fort Calhoun		X			11:05	7
"		X		X	:54	7
"		O			<:01	7
Ginna		O			:30	2
Ginna	X				:40	2
Haddam		O			:29	2
"		O			:09	2
"		O			<:01	2
"	O				:20	2
"		O			:16	2
Humbolt Bay		X*			----	2
Indian Point	X				6:28	2
La Crosse	O				:14	2
"		X			1:01	2
"		O			:20	2
"		O			:02	2
"		X		X	1:50	2
"		O			:10	2
"		X			5:35	2
Millstone 1		X		X	24:37	2
Nine Mile Point		O	X	X	<:01	2
Oconee		X			1:00	2
Oyster Creek		X*	X*	X*	----	3

See notes on following page.

Table D-1 (Continued)
CLASSIFICATION OF LOSP EVENTS APPEARING IN EPRI NP-2301

NUCLEAR PLANT (1)	OFF-SITE: GRID AND TRANSMISSION LINES (2)	MAIN SWITCH-YARD (3)	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS (4) (A)	(B)	RECOVERY TIME (Hrs:Min) (5)	NUMBER OF OFF-SITE TRANS-MISSION LINES (6)
Palisades		X			:56	5
"		X		X	4:45	5
"		X		X	3:30	5
"		X		X	1:30	5
Pilgrim	X				2:40	3
"	X				8:54	3
Point Beach		0			:08	2
"		X			:55	2
Quad Cities		X			1:11	4
San Onofre		X	X	X	4:59	7
"		0			:04	7
"		0			<:01	7
St. Lucie	0				:08	2
Yankee Rowe	X				:37	2
The total number of LOSP events is 45						

X - Indicates an event lasting >30 minutes.
0 - Indicates an event lasting <30 minutes.
* - Assumed, based on the specific failure mode.
Included in the calculations as a failure lasting >4 hours.

The recovery factors determined for LOSP events in this analysis incorporate some subtle but potentially important distinctions regarding the source of the data, i.e., the data set is composed of a spectrum of failures from rather minor single failure events through severe weather conditions affecting multiple transmission lines. Based upon the data in EPRI NP-2301 (causes and durations), it is clear that there is a strong coupling between the duration of the loss of off-site power outage and the particular failure mode. However, in the assessment of the on-site gas turbine capability the use of the data to allocate exact failure modes extends the statistical significance of the data to its limits. For this reason, this analysis attempts to examine the general trend of the coupling of the failure modes (e.g. weather) and

duration of LOSP and to provide both optimistic (CASE A) and pessismistic (CASE B) assumptions in interpretation of the data to account for uncertainties in classification of event. From this perspective, long duration transmission line failures take on an increasing importance. The analysis incorporates these data directly into the probabilistic evaluation including the strong coupling between failure mode and duration of LOSP.

Table D-2 summarizes the data and classifies the failure modes by time phase consistent with the PRA. The information in the table may be used to derive estimates for recovery values used in the recovery logic model.

The time phases which have the highest contribution to core vulnerable frequency following LOSP initiators, i.e., time phases III and IV, have a substantial fraction of events for which off-site power has not been recovered based upon historical nuclear plant operating experience.

Table D-2

SUMMARY OF LOSP EVENTS BY TIME PHASE

TIME PHASE	GRID AND OFF-SITE TRANS-MISSION	MAIN SWITCHYARD	POTENTIAL IMPACT ON REDUNDANT SWITCHYARDS	
			CASE A (optimistic)	CASE B (pessimistic)
PHASE I > 30 MIN	6	19	3	12
PHASE II > 2 HR	4	11	3	9
PHASE III > 4 HR	3	10	3	8
PHASE IV > 10 HR	1	2	1	1

REFERENCES

1. Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301, March, 1982.

2. Scholl, R.F., "Loss of Off-Site Power Survey Status Repor", Revision 3, Report of the Systematic Evaluation of Program Brench, Division of Licensing, U.S. NRC.

APPENDIX E

SENSITIVITY STUDY OF THE CORE VULNERABLE FREQUENCY ASSOCIATED WITH LOW POWER TESTING

The Main Report provides an indication of the assumed plant configuration for low power testing. For simplification, steady state operation at 5% power is assumed. In reality, however, the plant will be in a state of flux as numerous system tests, inspections, and measurements take place. The assumption of a steady state power level of 5% is judged to be a conservative approach to the assessment of risk for low power operation. The purpose of this Appendix is to provide additional bases for concluding that this approach is conservative. Specifically, a summary discussion of the detailed operational aspects of low power testing [1] is provided along with a best estimate, rather than conservative upper bound, evaluation of the core vulnerable frequency.

E.1 DETAILED OPERATIONAL ASPECTS OF LOW POWER TESTING

The initial plant testing at Shoreham involves a series of four phases defined as follows:

Phase I:	Fuel loading and precriticality testing
Phase II:	Cold criticality testing
Phase III:	Heatup and low power testing to rated pressure/temperature conditions (approximately 1% rated power)
Phase IV:	Low power testing (1-5% rated power)

The structure of the testing program is such that the plant must fulfill specific testing and operational objectives in each phase before continuing to the next. Component "wear-in" failures will be repaired as they arise, which implies that the plant will have a full complement of safety systems available for each new power level.

In this evaluation, it is judged that essentially no measurable risk can be associated with the first two phases of operation. This conclusion is based on two observations:

1. WASH-1400 [2] identified spent fuel handling accidents as a potential radionuclide release mechanism. Damage to unirradiated fuel would not fall into this category, however. Fresh fuel bundles can be handled in open air and would not be subject to melting or significant radionuclide release.

2. Cold criticality testing produces a negligible amount of heat (0.0001% to 0.001% of rated thermal power, or a maximum of 24kw). Under these conditions, there is essentially no need for RPV inventory makeup systems. If the RPV is inadvertantly drained, the core will become subcritical. Therefore, postulated precursors to core damage (e.g., LOCA or failure to scram) would have a negligible impact on core integrity.

Thus, Phases III and IV, in which the RPV is pressurized, steam is generated, and coolant makeup systems are required is judged to be the region of operation in which the first measurable risk due to radionuclide release appears. The testing program for these phases consists of repeated gradual heatups to a maximum of 5% full power followed by controlled cooldowns. Only a small fraction of the total operating period is spent near 5% power. Because of these power cycles, it is judged that decay heat levels, which are dependent upon the power history, can best be modeled using an intermediate valve between 0% and 5% and assuming steady state operation. For this reason, the core vulnerable frequency associated with operation at 2.5% power is estimated in the following section.

E.2 CORE VULNERABLE FREQUENCY DUE TO STEADY STATE OPERATION AT 2.5% POWER

As with the analysis presented in the Main Report, phenomenological calculations of postulated accident progressions are an essential starting point for assessing success criteria and quantifying event probabilities. The analysis presented in this section is intended to (1) review the results of MARCH calculations for accidents initiated for 2.5% power (2) identify and reassess events which are significantly different in timing or magnitude from the 5% power case; and (3) quantify these differences to estimate the core vulnerable frequency. This approach is judged to be the most effective method of

evaluating the sensitivity of the assumed power level. The results will be directly comparable to accident frequency estimates derived in the Main Report (5% power) and in the PRA (100% power).

E.2.1 ACCIDENT PROCESS CALCULATIONS

Appendix B presents the results of process calculations from MARCH for accidents postulated during low power operation. The emphasis is primarily on the evaluation of parameters assumed for steady state operation at 5% power. In particular, the differences between 5% and full power include an upwardly skewed axial power profile, an average core temperature of 640°F (representing a reduction of 400°F), and a reduced reactor pressure of 950 psia. Given these differences, the key modeling difference between the 5% and 2.5% power level calculations is the reduction in initial and decay heat power levels. With this in mind, Table 1 presents a summary of the estimated timing of postulated accident sequences in which the core becomes vulnerable to melting.

TABLE E-1

ACCIDENT SEQUENCE TIMING SUMMARY

SEQUENCE	TOTAL TIME TO CORE VULNERABILITY (HRS.)	
	5% POWER	2.5% POWER
Scram, Isolation, Failure of Coolant Injection	30	71 - 80
Scram, Large LOCA, Failure of Coolant Injection	3	10
ATWS, Containment Initially Intact	8 - 10	14 - 15

These timing estimates are judged to have a significant impact on the plant system success criteria beyond that already included in the Main Report. Three principal effects are as follows:

1. CRD pump flow is considered a viable alternative coolant injection source in accidents accelerated by a loss of coolant inventory. This applies to SORV cases, medium LOCAs, and large LOCAs, but not RPV ruptures.

2. RCIC alone is a viable alternative for coolant injection during an ATWS (at 5% power, a combination of RCIC and CRD flow is judged necessary).

3. The reactor power level following ATWS with RPT is estimated to be within the capacity of 1 RHR heat exchanger. This is based on the assumption that the power level will decrease by approximately 40% following RPT. Even if a single loop of RHR failed to match the ATWS power level, the challenge to containment is expected to be substantially extended. In these instances, a rationale similar to that described in the Main Report for dismissal of Class II challenges is judged applicable, i.e., the very long period of time available prior to containment failure represents a risk below that which can be credibly quantified. Therefore, such cases are judged to have a negligible frequency.

These success criteria are incorporated into the quantification of core vulnerable accident sequence frequencies in the remainder of this section. Additionally, revised event success criteria due to the timing estimates will be discussed as they arise.

E.2.2 QUANTIFICATION OF ACCIDENT SEQUENCE FREQUENCIES

This section corresponds to Section 3 of the Main Report. As such, a discussion and quantification of each of the four initiator types is presented; based on the revised success criteria for operation at 2.5% power.

E.2.1.1 LOSS OF OFFSITE POWER INITIATOR

The LOSP tree shown in Figure 3.1 of the Main Report consists of an initial subtree used to define groups of sequences with similar timing, followed by subsequent subtrees used for modeling time dependent events. For the reassessment at 2.5% power, the only event requantified is Event R: Recovery of Offsite Power. Table 3-1 presents a comparison of the estimated recovery probabilities for the 5% and 2.5% power cases. This requantification results in a reduced core vulnerable frequency estimate of 7.7E-7 events/reactor-year for LOSP initiators.

TABLE 3-1 CONDITIONAL PROBABILITY OF RECOVERY OF OFFSITE POWER AS A TIME DEPENDENT FUNCTION

ACCIDENT SEQUENCES	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (5% POWER)	CONDITIONAL PROBABILITY OF FAILURE TO RECOVER OFFSITE POWER (2.5% POWER)
TYPES		
1)	1.E-4++	1.E-4++
1)'	1.E-3++	1.E-4++
2)	5.E-3	1.E-4++
3)	.25	.06
4)	.06	.02
5)	0.13	.03

\+ Based on containment conditions

++ Estimates of recovery probability at times greater than 24 hours are based upon engineering judgement since insufficient data exists to characterize such recovery probabilities even on a generic basis.

E.2.2.2 LOSS OF COOLANT ACCIDENTS (LOCAs)

At 2.5% power, the viability of CRD pump flow has its principal impact on mitigation of large and medium LOCAs sequences involving failures of low pressure systems. Using an estimated CRD injection reliability of 0.9/demand (similar to the small LOCA case in the Main Report), the estimated frequency of core vulnerable conditions is reduced to 9.4E-8 events/reactor year and 6.0E-8

events/reactor-year for large and medium LOCAs, respectively. Neither the additional time available nor the reduced challenge to plant systems from that described for 5% power is judged to have a quantifiable impact on other LOCA initiated core vulnerable sequences. Therefore, the total estimated frequency of core vulnerable conditions due to LOCAs initiated from 2.5% power is reduced to 2.2E-7 events/reactor-year.

E.2.2.3 OTHER TRANSIENTS

A significant contributor to the frequency of accidents initiated by "other" transients at 5% power is the failure to maintain primary system integrity. Transient induced LOCAs or SORVs tend to accelerate core heatup timing because coolant is lost early in the sequence. At 2.5% power, CRD flow is considered a viable alternative for coolant injection in these cases. Assuming credit for CRD flow similar to other accident sequences in this category at 5% power (i.e., a reliability of 0.99/demand), then the frequency estimates for induced LOCA or SORV accident sequences are reduced by two orders of magnitude.

Other dominant contributors to the frequency of core vulnerable conditions are postulated to involve failures of depressurization systems which prevents injection by low pressure systems. Intuitively, the extended core heatup timing provides a basis for arguing that failures of depressurization systems may be recovered prior to reaching unacceptably high fuel temperatures. However, the data and modeling required to support this assertion is a level of effort beyond the scope of this analysis. Therefore, it is judged that the already high combined reliability of depressurization systems at 5% power is adequate for the estimated reliability at 2.5% power. Thus, the total core vulnerable frequency due to other transients at 2.5% power is estimated to be 3.5E-7 events/reactor-year.

E.2.2.4 ATWS

The quantification of ATWS event trees at 5% power in the Main Report includes several changes in success criteria based on sequence timing. The differences between 5% and 2.5% power are judged to be negligible with regard to the

requantification of events appearing in ATWS sequences. Therefore, the estimated frequency of core vulnerable conditions is the same for both ones.

E.2.3 COMPARISON AND SUMMARY

Table 3-2 summarizes the quantification of accident sequences frequencies at 2.5% power. As shown, the total frequency of core vulnerable conditions is reduced by approximately a factor of 3, primarily due to the extended timing of LOSP sequences. An additional contributor is the assumed viability of CRD injection as a means of core cooling, which is also attributed to the extended sequence timing. While not explicitly calculated, it is judged that the extended sequence timing would have a significant favorable impact on other parameters important to risk calculations including: evacuation warning times, in-containment residence times, etc.

TABLE 3-2

DOMINANT ACCIDENT SEQUENCE FREQUENCIES ASSUMING STEADY STATE OPERATION AT 2.5% POWER

INITIATOR TYPE	TOTAL SEQUENCE FREQUENCY
Loss of Offsite Power	7.7E-7
LOCAs	2.2E-7
Other Transients	3.5E-7
ATWS	2.7E-7
TOTAL	1.6E-6

REFERENCES

1. Supplemental Motion for Low Power Operating License, submitted in the matter of Long Island Lighting Company, Docket No. 50-322, to the Atomic Safety and Licensing Board, affadavit of J. A. Notaro and W. E. Gunther, Jr., dated March 20, 1984.

2. Reactor Safety Study, WASH-1400, NUREG75/014, dated October 1975.