NUREG-1068

Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

In recognition of the high population density around the Limerick Generating Station site and the proposed power level, the Philadelphia Electric Company, in response to NRC staff requests, conducted and submitted between March 1981 and November 1983 a probabilistic risk assessment (PRA) on internal event contributors and a severe accident risk assessment on external event contributors to assess risks posed by operation of the plant. The applicant has developed perspectives using PRA models on the safety profile of the Limerick plant and has altered the plant design to reduce accident vulnerabilities identified in these PRAs. The staff's review of the Limerick PRA has particularly emphasized the dominant accident sequences and the resulting insights into demonstration of compliance with regulatory requirements, unique design features and major plant vulnerabilities to assess the need for any additional measures to further improve the safety of the LGS. The staff's review insights and PRA safety review conclusions are presented in this report.

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ABBREVIATIONS

ADS	automatic depressurization system
ANSI	American National Standards Institute
ARC	alternate room cooling
ARI	alternate rod insertion
ASLB	Atomic Safety Licensing Board
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling-water reactor
COR	containment overpressure relief
CS	containment spray
CST	condensate storage tank
DC	U.S. Department of Commerce
DES	Draft Environmental Statement
ECC	emergency core cooling
EERI	Earthquake Engineering Research Institute
EQE	EQE Incorporated
ESW	emergency service water
FES	Final Environmental Statement
FW	feedwater
GE	General Electric Company
HPCI	high pressure coolant injection
IE	Office of Inspection and Enforcement, NRC
IEEE	Institute of Electrical and Electronics Engineers
ILRT	integrated leak rate testing
INEL	Idaho National Engineering Laboratory
JAERI	Japan Atomic Energy Research Institute
LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
LPCS	low pressure core spray
MSIV	main steam isolation valve
NAS	National Academy of Sciences
NCEE	U.S. National Conference on Earthquake Engineering
NCEL	U.S. Naval Civil Engineering Laboratory

ABBREVIATIONS (Continued)

NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PECO	Philadelphia Electric Company
PRA	probabilistic risk assessment
QA	quality assurance
RB	reactor building
RCIC	reactor core isolation cooling
RES	Richardson Engineering Services, Inc.
RHR	residual heat removal
RPT	recirculation pump trip
RPV	reactor pressure vessel
RSS	Reactor Safety Study (WASH-1400)
SARA	Severe Accident Risk Assessment (also called PRA)
SGTS	standby gas treatment system
SLC	standby liquid control
SW	service water
SWRI	Southwest Research Institute
TRIP	transient response implementation procedure

SYMBOLS

Q	failure of feedwater system
T_E	loss of offsite AC Power
T_F	MSIV closure
T_T	turbine trip
T_W	total loss of containment heat removal function
U	failure of high pressure coolant systems
V	failure of low pressure coolant systems
X	human failure rate to depressurize the reactor

REVIEW INSIGHTS ON THE PRA FOR THE LIMERICK GENERATING STATION

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF REVIEW

1.1 INTRODUCTION

This report discusses the NRC staff's evaluation of the probabilistic risk assessments conducted by the Philadelphia Electric Company for the Limerick Generating Station. The site is located on the Schuylkill River about 1.7 miles southeast of the limits of the Borough of Pottstown in Montgomery and Chester Counties, Pennsylvania. The site is about 21 miles northwest of the Philadelphia city limits.

The Limerick Generating Station uses a BWR 4 boiling water reactor (BWR) designed and supplied by the General Electric Company (GE). It is similar to other BWRs such as Susquehanna, Zimmer 1, and Peach Bottom. The reactor consists of a reactor pressure vessel that contains a core, control rods, instrumentation, steam separator and dryer assemblies, jet pumps, and control rod drive mechanisms. The core contains 764 fuel assemblies and 185 movable control rods arranged in an upright circular cylindrical configuration. The design power level of each reactor is 3435 megawatts thermal (MWt). The rated power level is 3293 MWt per unit. The design gross electrical output is 1138 MWe per unit.

The reactor system is housed in a GE Mark II containment. The containment systems include a pressure suppression containment structure (primary containment), the secondary containment structure and supporting systems, the containment heat removal system, the containment isolation system, and the combustible gas control system. The primary containment is in the form of a truncated cone over a cylindrical section, with the drywell being the upper conical section and the suppression chamber being the lower cylindrical section. These two sections comprise a structually integrated, reinforced concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. The drywell and suppression chamber are divided by a horizontal diaphram slab of reinforced concrete structurally connected to the containment wall. The drywell encloses the reactor vessel, the reactor recirculation system, and the associated piping and valves. The suppression chamber (wetwell) consists of an air region and a water region (suppression pool). The primary and secondary containment structures and associated systems function to prevent or control the release of radioactive material that might be released into the containment atmosphere following a postulated accident.

The reactor protection system is designed to initiate a rapid, automatic reactor shutdown (scram) if selected monitored system variables exceed preestablished limits. Four emergency core cooling systems are designed to prevent core overheating in the event of a postulated loss-of-coolant accident. These systems include the high pressure coolant injection system, automatic depressurization system, low pressure core spray system, and the low pressure coolant injection system (an operating mode of the residual heat removal system). Offsite ac power for Units 1 and 2 is provided by two 230-kV transmission lines and two 500-kV transmission lines, with a third 500-kV line to be added. Four diesel generators per unit provide standby onsite ac power. The plant dc system will provide power to vital instrumentation and controls if normal ac plant power sources become unavailable.

The Philadelphia Electric Company (PECO) is the applicant for the operating licenses for the Limerick Generating Station. Bechtel Power Corporation is providing architectural, engineering, construction and startup services. GE designed, fabricated and delivered to the site the nuclear steam supply system, the fuel and the turbine generators.

This report summarizes the results of the NRC staff's review of the applicant's probabilistic risk assessments (PRA) as they relate to the safe operation of the plant. The report presents the staff's safety review insights with particular emphasis on dominant sequences and the resulting insights into regulatory compliances, unique design features, the applicant's voluntary safety improvements, limited comparison of Limerick with other plants and potential plant vulnerabilities to assess the need for any additional measures. The review of the PRAs was conducted in accordance with the guidance available to the NRC staff which includes the Commission's Policy Statement of January 18, 1979, Staff Actions Regarding Risk Assessment Review Group Report, the Commission's Policy Statement on Safety Goals, 48 Fed Reg 10772, March 14, 1983, and the Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, 48 Fed Reg 16014, April 13, 1983.

The results of the NRC staff's review of the Limerick operating license application relative to the radiological safety review requirements of Title 10 of the Code of Federal Regulations are reported elsewhere in the Safety Evaluation Report (NUREG-0991) and its supplements. The results of the NRC staff's assessment of the environmental impact associated with the operation of Limerick pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51 are reported elsewhere in the NRC staff's Draft Environmental Statement (NUREG-0974), Supplement No. 1 to the DES and in the Final Environmental Statement.

During the course of its review the staff held a number of meetings with representatives of the applicant. The staff requested additional information which the applicant provided in five revisions to the initially submitted internal events PRA and in two revisions to the external events severe accidents risk assessment report. This information is available to the public for review at the NRC Public Document Room at 1717 H Street, NW, Washington, DC and at the Local Public Document Room at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Consistent with the above considerations, this report discusses the major safety-related insights and conclusions reached by the staff during its review of the Limerick internal events PRA and the external events severe accident risk assessment. (Hereafter the term PRA refers to both of these reports collectively unless otherwise noted). Section 2 discusses the overall insights obtained from the risk review. Section 3 discusses the insights on dominant sequences and the applicant's compliance with the NRC's deterministic regulatory requirements. Section 4 discusses the staff's review of external events. Section 5 discusses voluntary plant design changes made by the applicant which are believed to have been influenced by the PRA results. Section 6 discusses some perspectives on comparison of Limerick with other plants. Section 7 discusses the adequacy of the Mark II containment to mitigate core damage accidents. Section 8 discusses some important engineering and operational insights that should further improve safety in a cost-effective manner.

1.2 High Population Density Considerations

In early 1979, a staff report on population near nuclear sites, NUREG-0348, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites" (NRC, November 1979) documented the population distribution around all U.S. nuclear power stations within circles of various radii, such as 10, 30 and 50 miles, on the basis of the 1970 census. This report indicated that the region around Limerick was one of the three most densely populated site areas in the United States. In particular, the 1970 census indicated that the population statistics of the Limerick site were 152,644 persons within a 10-mile circle and 7,036,199 within a 50-mile circle. Due to a combination of factors which included the high population densities and the proposed power levels, the staff considered at that time (1979-1980) that Limerick might be one of the plants whose operation could represent a disproportionately high segment of the total societal risk from reactor accidents. Therefore, in May 1980, the staff requested that PECO perform a preliminary risk assessment and compare that risk to the risk identified with the Reactor Safety Study (RSS) WASH-1400 reference BWR plant, as appropriate. The staff also discussed its request of PECO to perform a PRA and its outline for the review of the PRA in SECY 81-25, "Performance of Probabilitistic Risk Assessment or Other Types of Special Analyses at High Population Density Sites" on January 12, 1981.

The staff requested that PECo conduct its PRA "utilizing the WASH-1400 methodology, but taking into account significant differences between the WASH-1400 reference plant and the Limerick facility." Also PECO was asked to account for plant design differences and site-specific differences between Limerick and the RSS BWR plant. In response PECO performed a PRA with the help of its consultants (General Electric, Science Applications Inc., Henry and Fauskee Associates and the Bechtel Power Corporation) which analyzed the contribution of internally initiated events. This report was submitted in March 1981. In April 1983 the applicant submitted a severe accident risk assessment report which added to the earlier report by including an analysis of the contribution of externally initiated events.

1.3 Conduct of Review

The staff initiated the bulk of its review of the PRA in February 1982. As part of the staff's review, the Brookhaven National Laboratory (BNL) was contracted to review the Limerick PRA. The major review objectives were to:

(1) Identify the dominant accident sequences,
(2) Identify major risk contributors, and
(3) Compare the risk posed by Limerick to the RSS BWR.

These review objectives were established at that time to determine whether operation of Limerick would present a disproportionately high segment of the total societal risk posed by the operation of all power reactors licensed to operate by the NRC. If the staff were to find evidence of disproportionate risk, then the PRA and the staff's review of it consistent with the above objectives would also facilitate the staff's decision to determine whether any additional design features or other actions would be necessary.

The staff, PECO and consultants for both interacted frequently in the risk review process. Several public review meetings were held. PECO presented overviews of the PRA to the staff and BNL first on February 11-12, 1982 and again on March 1, 1982. In response to the staff's requests for additional information, the applicant provided the staff with five revisions to the Limerick PRA during the period between March 1981 and October 12, 1982. The applicant's PRA and its five revisions were based on the plant design configurations such as human-dependant ADS (automatic depressurization system) initiation logic, as documented in Revision 18 of the Limerick Final Safety Analysis Report (PECO, March 1983) and were based on operating procedures and Technical Specifications of Peach Bottom Nuclear Station operated by the same applicant. The applicant was very cooperative in providing the needed information for the staff's review of the PRA. On the basis of the information provided in these documents and with clarifications provided at meetings with PECO and during site visits, BNL audited the PRA in specific areas to test the results against alternative assumptions and supplemental data. BNL completed its review and NRC first issued the results of that review as a draft report (NUREG/CR-3028) on October 27, 1982, as Board Notification No. 82-108. During the ensuing review, technical comments were sent to BNL from the NRR staff and the applicant. BNL incorporated these comments, as appropriate, and submitted its final report (NUREG/CR-3028) to the staff which was issued as Board Notification No. 83-25 on March 4, 1983.

On April 21, 1983 PECO submitted a Severe Accident Risk Assessment (SARA) report which added to the earlier report by including a risk assessment of external hazards such as earthquakes, fires, floods, tornadoes, turbine missiles and also an analysis of random reactor vessel failure. The staff and BNL reviewed this additional risk assessment including two revisions provided in July and November 1983. BNL's review of the earthquake and fire events are reported in NUREG/CR-3493 and in BNL 33835. Although the staff's review of the external hazards risk assessment was performed subsequent to the review of the internal hazards risks assessment, both are reported herein.

1.4 Use of PRA

Some aspects of the staff's views on the use of the Limerick PRAs have evolved somewhat since the applicant was first requested to perform a PRA as the state of the art of PRA knowledge has advanced.

The staff's initial expression on the usage of the Limerick PRA was set forth in the staff's May 6, 1980 letter to the applicant as discussed in Section 1.2. The staff visited the subject of usage of the PRA again in an affadavit filed with the Atomic Safety and Licensing Board (ASLB) for Limerick on February 1, 1982 in response to an ASLB request. The staff again responded to an ASLB request by filing on April 13, 1983 and May 24, 1983 additional statements on the NRC staff's use of the Limerick PRA in the safety portion of the Limerick proceeding, in the environmental portion of the Limerick proceeding and for additional uses outside the Limerick licensing proceeding. These uses are summarized below.

Use in Safety Portion Of Limerick Proceeding

The Staff used the information that evolved from the review of the Limerick PRA, particularly information concerning risk dominant sequences, to check whether such sequences were attributable to structures, systems, components or procedures which failed to satisfy NRC regulatory requirements. If non-conformances had been identified, then the items involved would have been changed to conform to NRC requirements in order for the necessary licensing findings to be made.

In the event that a dominant risk sequence had been identified which was significant to overall facility safety but was attributable not to a failure of compliance with Commission regulations but to a unique design aspect of Limerick, the Staff would have considered additional measures to compensate for the unique problem.[1]

To the extent that such information has some significant relationship to the Limerick design, the Staff has used information relating to such matters as potentially significant sequences, specific system or component failure data, and containment failure models as derived from its review of other PRA's to test the reasonableness of data and assumptions used in and conclusions resulting from the Limerick PRA. The PRA review supplemented the staff's traditional deterministic safety review.

The staff's Regional Office also some plans usage of the PRA as discussed in a letter to the applicant dated February 28, 1984 to provide a priority ranking of the relative importance to safety of systems and components.

Use in Environmental Portion of Limerick Proceeding

The staff planned to use the information resulting from its review of the Limerick PRA to assess the risk of accidents beyond the design bases, in accordance with the Commission's Statement of Interim Policy Concerning Nuclear Power Plant Accident Consideration Under the National Environmental Policy Act of 1969, 45 Fed. Reg. 40101 (July 13, 1980). The discussion of accidents beyond the design bases in the Limerick Environmental Statements (DES and FES) was along the general lines of such discussions in other Environmental Statements issued since the Commission published its Interim Policy Statement. However, the underlying information in the Limerick DES and FES was case-speciic data derived from the Limerick PRA where other recent FES's have used generic information adjusted to the specific case.

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1 Depending on the nature of such unique problem, if any, there are various regulatory provisions which may be applicable: e.g. the implementation of 10 CFR Part 100 has included consideration of compensatory engineered safety features to offset adverse siting characteristics such as large nearby populations.

The staff planned to compare the overall risk of Limerick with the overall risk of other facilities using such information as may be available from other PRA's, including WASH-1400. From this, the Staff planned to assess whether the risk at Limerick is significantly greater than that associated with other reactor facilities in general, giving due consideration to the wide range of uncertainties that may be involved.

If the risk were determined to be within the range associated with other facilities, then the comparison would be of only background use in the Limerick proceeding (that is, in providing perspective on the fact that Limerick has no unusual characteristics). The staff planned a comparison of offsite risks associated with plant accidents, with the risks of normal operation and with the risks associated with other human activities in the area surrounding the Limerick site. The ultimate comparison of significance is whether the environmental impacts of Limerick (including this impact) are outweighed by the benefits of Limerick.

If the risk were determined to be significantly greater than that associated with other reactor facilities, (that is, to be disproportionate), then the Staff would have considered the need to recommend compensatory features.

Additional Uses Outside The Limerick Licensing Proceeding

In addition to the uses to which the Staff put the Limerick PRA in the licensing of Limerick, the Staff is using the Limerick PRA as a part of its general expansion of the scope of PRA knowledge and as a potential source of information concerning safety effectiveness and costs of prevention and mitigation features for the severe accident rulemaking program for proposed new standard plants or other possible severe accident rulemaking activities.

Another purpose for which the PRA was used was as a basis for voluntary improvements in the facility. In fact, the applicant has already used the Limerick PRA as a basis for making voluntary improvements at Limerick.[2]

The Commission has recently stated in its policy statement, Safety Goal Development Program, 48 Fed. Reg. 10772 (March 14, 1983), that the safety goals "will not replace the NRC's reactor regulations [and that the] NRC will continue to use conformance to the regulatory requirements as the exclusive licensing basis for plants." We believe that the Staff's use of the PRA as reflected above is consistent with the Commission's safety goals policy statement.

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[2] These include: (a) redundant air supplies to the ADS, (b) separate injection nozzles for coolant makeup into the reactor vessel, (c) added crossover valves in the RHR service water systems, and (d) new procedures to enhance the recovery of the power conversion system for containment heat removal. In addition, the Applicant's choice of the ATWS prevention/mitigation fix referred to as Option 3A was aided by information derived from the PRA.

1.5 Conclusions

In recognition of the high population density around the Limerick site and the proposed power level, the applicant has performed a full-scope PRA and has developed perspectives on the safety profile of the Limerick plant. The applicant has identified and evaluated, through the use of the PRA models, some major plant improvements such as the ATWS alternate 3A modifications and the residual heat removal (RHR) system modifications which have reduced the risk at Limerick significantly. The staff has completed its review and reports the detailed findings in Section 2 through 8 of this report. The staff's conclusions are summarized as follows.

1.5.1 The Safety Review

Both the Limerick PRA and SARA are at the current state of the art in risk assessments. The staff has reached the following major conclusions concerning the safety review of Limerick.

(a) The review of the dominant accident sequences from the probabilistic safety assessments has found no instances of noncompliance with the deterministic regulatory requirements.

(b) Limerick does not have unique design features which are significant with respect to the plant's susceptibility to the dominant sequences.

(c) The interactive nature of the review led the utility to revise the PRA five times and the SARA twice. The utility established that a substantial safety improvement in a dominant sequence was obtained by design changes to enhance the redundancy of the RHR service water supply. Also, the utility achieved a significant reduction in the estimated frequency of core damage and the potential for early fatalities by increasing the redundancy and reliability of the standby liquid control system. Several other design modifications influenced by the PRA were also implemented. The PRA supplemented the analytical basis for the utility to decide on these design improvements.

(d) Because of differences in scope, data, methods and assumptions, a systematic comparison of risk results from various PRAs is not possible at this stage. However, the Limerick design includes sufficient risk-reducing improvements that, despite the high population density at this site, the best estimates indicate that the calculated risk from operation of this facility is not significantly different from the risks reported for the Indian Point 2, 3 and Zion nuclear facilities. For example, with the ATWS 3A modifications, the operation of Limerick does not seem to pose a disproportionate share of the societal risk compared with the Indian Point Units 2 and 3 and Zion plants which are also located in areas of high population density.

(e) The dominant contributors to the core damage frequency are the transient and LOCA events. External events collectively contribute less than 25% of the core damage frequency at Limerick. However there are considerable uncertainties associated with such comparisons and much emphasis should not be placed upon them at this time. The risk assessments provide evidence that the Limerick plant is capable of withstanding external events beyond the design basis events.

(f) The PRA provided evidence that three additional improvements could reduce the vulnerability of Limerick to core damage accidents. First, the procedures reviewed during the initial PRA review indicated that the decisions regarding depressurization of the reactor during some transients could be improved by upgrading the emergency procedures. Second, the availability of the high pressure coolant injection systems could be increased by providing an alternative means, e.g., additional batteries, battery powered fans, for ambient room cooling. Third the containment failure time could be increased and thus core damage risk could be reduced by increasing the availability of the containment spray system. These issues were communicated to the applicant in a meeting held on March 30, 1984 and the applicant responded to them in a letter dated April 11, 1984. The staff believes that the applicant has provided a reasonable and sufficient response to these three issues. A more detailed discussion of them may be found in Section 8.

(g) In recognition of the substantial uncertainties in the probabilistic safety assessments around Limerick's already high population density site, it appears reasonable and prudent that the applicant establish and implement a safety assurance program with the objectives of assuring that the conduct of operations and future clues to the safety of the plants are and remain consistent with a level of severe accident risk not appreciably greater than that assessed by the staff. The safety assurance program is discussed in Section 8.2 and in Appendix B.

In summary considering the plant design features and additional safety improvements in conjunction with the review of the Limerick PRA and the SARA, the staff believes that the operation of the Limerick Generation Station will pose no undue risk to the public health and safety.

1.5.2 The Environmental Review

The Limerick probabilistic safety studies provided plant-specific accident sequences and risk consequences that are more representative of Limerick than are those of the surrogate plant in WASH-1400. Thus, the staff used these studies in Supplement 1 to the Draft Environmental Statement (NUREG-0974, December 1983) and the Final Environmental Statement (NUREG-0974, April 1984) for Limerick.

1.5.3 Severe Accident Decisionmaking Activities

The staff believes that the applicant has made substantial plant design improvements to correct the vulnerabilities of the sequences involving severe releases at Limerick and, therefore, has reduced risk at Limerick significantly. Thus, the staff finds no basis, at this time, to conclude that special mitigation systems to reduce risk are warranted at Limerick.

Finally, the staff will use both the Limerick PRA and the SARA as sources of information typical of a BWR Mark II design in its generic severe accident decisionmaking activities.

2.0 OVERALL PRA REVIEW INSIGHTS

2.1 Accident Sequence Analysis

The Limerick risk study has made use of RSS-type event tree techniques and has developed the potential sequences of core damage accidents, depending on the failure of the basic safety functions such as loss of coolant inventory makeup, loss of containment heat removal, and loss of scram. Overall, the Limerick PRA has identified 86 core damage sequences. To facilitate the analyses of physical processes and containment failure modes, core damage sequences have been grouped into various classes which are based on the safety function failed and the containment condition. Six classes of core damage accident sequences are assigned in the Limerick PRA. The characteristics of these six classes, which are also discussed in BNL-33835, follow.

Class I consists of sequences in which a transient is followed by a loss of reactor coolant inventory, leading to core melt in an intact reactor vessel. Radionuclides released while the core is melting pass through the safety relief valves into the suppression pool, which is subcooled in these sequences and provides efficient scrubbing. The core melts and slumps to the bottom of the vessel, which rapidly fails, whereupon the molten core drops to the diaphragm floor. Core-concrete interactions generate noncondensible gases, which ultimately cause containment failure.

Class II consists of sequences in which there is a failure of containment heat removal. Decay heat is rejected to the pool, which heats up and causes the containment pressure to rise until the containment fails. Containment failure then precipitates a loss of coolant inventory, with subsequent core melt and vessel failure. Radionuclides from the melt release pass through the safety relief valves into the suppression pool (if it is available), which is assumed to be saturated.

Class III consists of ATWS sequences with rapid core melt and vessel failure before containment failure. The sequence of events is much like that of Class I, except that the pool is assumed to be saturated.

Class IV consists of ATWS sequences with containment failure before core melt. Containment failure precipitates a loss of coolant makeup capability and subsequent rapid core melt and vessel failure. The radionuclides from the melt release then pass through the safety relief valves into the pool, which is assumed to be saturated.

Class IS consists of seismic sequences involving reactor enclosure failure. These sequences involve the failure of lateral supports of residual heat removal (RHR) heat exchangers, causing RHR suction lines to fail and drain out the wetwell below the level of downcomers. In terms of physical processes, these sequences are similar to Class I sequences except that the containment is predicted to fail early and the wetwell is predicted to be ineffective in the scrubbing function.

Class S consists of sequences in the failure of the reactor pressure vessel (RPV) itself followed by immediate containment failure. Vessel failure is followed by the rapid release of a large quantity of the pressurized water and steam. These sequences cover failures of a wide range of severity and are severe enough for the emergency core cooling (ECC) systems to fail in cooling the core.

The number of sequences that fall into each class of core damage sequences are listed below:

Class I: 23 sequences involving loss of RPV coolant inventory makeup induced by transients, fire and seismic events;

Class II: 15 sequences involving loss of containment heat removal induced by transients and seismic events;

Class III: 23 sequences involving loss of scram with inability to provide coolant makeup to RPV;

Class IV: 20 sequences involving loss of scram with inability to remove containment heat;

Class IS: 2 sequences involving simultaneous loss of RPV coolant inventory makeup and loss of wetwell water induced by seismic events; and

Class S: 3 sequences involving loss of the RPV itself.

Also, the Limerick PRA has quantified the 86 core damage sequences, using RSS type, but improved, fault tree techniques and has estimated the Limerick core damage frequency. A summary of the estimates of the frequency of core damage at Limerick and various events that contribute to the core damage, as estimated by PECO and staff review, is shown in Table 1. This table compares internally initiated events to externally initiated events. The staff is not aware of any systematic study into the differences in sources and treatment of uncertainty between internal versus external events. Thus caution should be exercised in comparisons that mix the different types of events.

Table 1 Frequency of Core Damage at Limerick

Contributors	PECO	Review
Transients and LOCAS	1.5E-5	8.5E-5 1/
Fires	3.4E-6	2/
Seismic events	5.7E-6	6/
Flood	3/	3/
Tornado	3/	2/, 4/
Turbine missiles	3/	3/
Random vessel failure	2.7E-8	2/
Total	2.4E-5/RY 5/	9E-5/RY

Table 1 Frequency of Core Damage at Limerick
(Continued)

1/Frequency increase is due to the added common mode failures and revised transient frequency.

2/Review indicates that PECO's frequency estimates seem reasonable. See Table 4 for uncertainties associated with these estimates.

3/Negligible (less than 1.0E-7 per reactor year).

4/PECO has submitted analyses to demonstrate that the ultimate heat sink piping can withstand tornado missiles with the criteria that the probability of exceeding 10 CFR 100 limits is less than 1.0E-7 per reactor year.

5/PECO performed ATWS, RHR, and fire-related fixes and reduced the total core damage frequency to 2.4E-5 per reactor year.

6/The staff's review did not provide a specific alternate estimate to that of PECO. See Table 4 for uncertainties associated with these estimates.

The staff has arrived at some insights as discussed below about the relative contribution of safety function failures and initiators to frequency of core damage.

When categorized by failure of a basic safety function, the failure to provide coolant makeup dominates the core damage frequency (about 90% contribution). The loss of long-term containment heat removal function and the loss of scram function make equal contributions, (about 5.0% each) to the total core damage frequency.

When categorized by examination of the dominant sequences the 86 total core damage sequences are found to be dominated by only 9 sequences, which contribute 90% to the total core damage frequency. Each of these 9 dominant sequences has an estimated frequency greater than 1E-6 per reactor year. Seven of these nine dominant sequences are induced by transients which contribute about 80% of the total core damage frequency. Within the class of transient initiators, vessel isolation transient and loss of offsite power contribute equally, about 40% each, to the total core damage frequency. The remaining contribution is made by transients such as turbine trip, inadvertently open relief valve, and manual shutdown. Other initiators such as LOCAs, fire, and seismic events contribute about 10% of the core damage frequency.

A summary of the first five dominant sequences and their frequency estimates is shown in Table 2. These five dominant sequences contributing about 77% of the total frequency of core damage, are initiated by transients followed by the failure of high pressure coolant systems such as high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) and the human failure to depressurize the reactor (x) in a timely fashion and common mode failure of the diesels. BNL reviewed and reassessed these dominant sequences. As part of its reassessment BNL identified 6 new sequences induced by support system failures. BNL added the effects of these failures of support systems such as DC power, AC power and service water to the system modeling and

Table 2 Dominant Accident Sequences at Limerick

Sequence	PECO	REVIEW	Comment on Differences
A loss of offsite power with a common cause failure of all diesel generators no timely recovery of AC power, and loss of inventory makeup systems (T_EUV)	5.9E-6	1.8E-5	Higher initiator freuquency, higher HPCI unavailability 1/
Reactor isolation with failure to restore the feedwater and condensate system, failure of higher pressure injection, and failure of timely ADS actuation (T_FQUX)	3.6E-6	3.7E-5	High T_FQ dependency, higher HPCI unavailability, higher human failure probability to depressurize 1/ 2/
Turbine trip followed by loss of feedwater, failure of higher pressure injection, and failure to depressurize the reactor (T_TQUX)	7.7E-7	8.0E-6	High turbine trip frequency, higher HPCI unavailability, higher human failure probability to depressurize 1/ 2/
Loss of offsite power with loss of high pressure injection due either to failure to restore AC power or due to random failures followed by failure to initiate ADS (T_EUX)	6.9E-7	5.0E-6	Higher initiator frequency, higher HPCI unavailability, higher human failure probability to depressurize 1/ 2/
Inadvertent opening of a relief valve followed by a failure of high-pressure injection and failure to initiate ADS (T_IUX)	6.8E-7	4.0E-6	Higher initiator frequency, higher HPCI unavailability, higher human failure probability to depressurize 1/ 2/

1/BNL has quantified the effect of support system dependencies (AC, DC, SW) at the accident sequence level. This contributed to increase in sequence frequency.

2/These values were determined prior to the implementation of TMI Action Plan Item II.K.3.18 regarding modifications to the actuation logic for ADS and, therefore, may not be fully representative of the current plant design.

requantified the effect of support system dependencies at the sequence level. BNL also reassessed the frequency of the transient initiators and the frequency of core damage. As a result of the above considerations, the core damage frequency of internal events was estimated to be 8.5E-5 per reactor year.

The frequency of random reactor vessel failure is estimated by the applicant to be about 2.7×10^{-7} per reactor year. This estimate is based on the same rationale as expressed in the Reactor Safety Study (RSS). The applicant has assigned ten percent of this frequency to the class of core damage accident sequences designated as Class S. The staff's risk assessment, as also discussed in the BNL-33835 report, was also based on the RSS median estimate of the gross vessel failure event and the assignment of a frequency value of 2.7×10^{-8} to the Class S sequences for this event. The applicant indicates that there is substantial uncertainty regarding the coolant boil-off rate for the blowdown and emergency core cooling refill rate for these events and has assigned fifty percent of the Class S frequency to the release category for which water is available in the vessel and the remaining frequency to the category for which water is not available. The staff considers the applicant's above source terms assumptions for random vessel failure sequences to be reasonable and has also used these assumptions in the staff's risk assessment.

BNL's review of risk assessment for external events was limited to only seismic events and fires, and the review included the state-of-the-art methodologies used for determining the event frequencies and their impact on critical structures and components. Particular attention was paid to the critical assumptions made in modeling component faults and underlying uncertainties. BNL also performed some requantification of dominant seismic and fire sequences in order to illustrate the sensitivities of the results to the underlying assumptions. Results of BNL's review of seismic events and fires are reported in NUREG/CR-3493. The staff's evaluation of seismic event frequencies and fragilities of critical components and structures are discussed in Appendix C. The BNL findings indicate that the applicant's estimate of seismic sequence frequencies seem reasonable. The staff findings indicate that the plant does not have any particularly weak links which would jeapardize protection against the seismic hazard. The review of dominant seismic sequences did indicate that the high pressure coolant systems may be tripped by the relay chattering in the isolation circuits at low g levels. However, as stated in the second revision to the SARA the applicant has included procedures to enable the operator to recover the tripped high pressure coolant systems by simple reset actions in the control room panels. The control circuits associated with circuit breakers for support system pumps, valves, and other instrumentation may also require resetting following the relay chatter at low "g" level. The applicant has addressed these issues in the revised transient response implementation procedure (TRIP). This appears to be a reasonable and sufficient response to this issue.

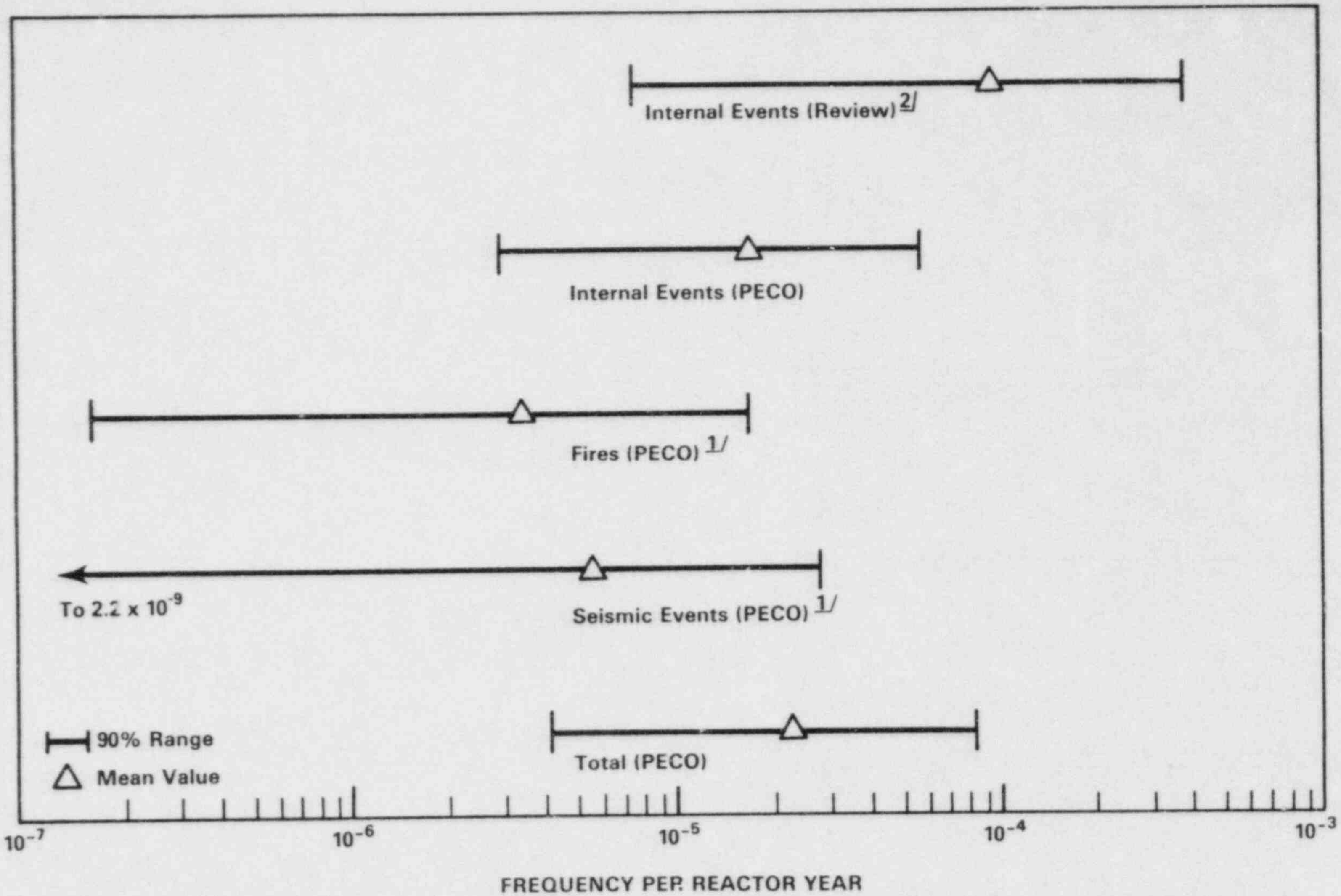
The staff and its consultant, BNL, have also reviewed the applicant's fire growth models, fire suppression models, and fire sequence frequency estimates and have reported the results in Section 4.2 and in NUREG/CR-3493. These reports indicate that the applicant has significantly reduced the fire sequence frequency by providing additional fire barriers for redundant shutdown systems (also see Revision 2 to SARA). The results of the staff's reviews of risk from floods, tornado missiles, and turbine missiles have also been reported in Section 4.0. These findings indicate that floods, turbine missiles, and tornado missiles do not contribute to the frequency of core damage by a significant amount (see Table 1).

2.2 Containment and Consequence Analysis

In order to obtain the consequences of various accident sequences, all 86 sequences were analyzed in the PRA to determine various containment failure modes and associated radioactive releases resulting from the core damage accidents. As part of the core melt progression and containment response analysis, the six classes of sequences discussed in Section 2.1 are combined with potential containment failure modes such as overpressure and leakage to obtain the appropriate release categories. The sequence class frequency was then multiplied by the conditional probability of the containment failure mode, and the resulting frequency was assigned to the proper release category. There are 11 release categories identified by PECO. For each of the 11 release categories, the source terms such as radioactive release fractions, delay time, energy release, and the height of release were estimated by PECO using RSS methodology assumptions on retention and holdup of radioactive chemicals. Then, the consequences, such as person-rems, early fatalities, and latent fatalities, were estimated by PECO for the Limerick site, using an appropriate evacuation scheme and the historical meteorological data. However, PECO's 11 release categories reflect the result of grouping several of the six individual classes of sequences and their source terms. In some of these categories the representative source terms had very dissimilar release characteristics and release fractions. Therefore the staff's evaluation involved grouping of various sequences into 27 release categories and involved estimates of the associated source term. The staff has also estimated early fatalities, latent cancers, and person-rems for each of the 27 release categories. Summaries of various release categories and estimates of the frequency and ranges of early and latent fatalities are shown in the Limerick FES and also in Figures 2 and 3. The risk estimates prepared by the staff and PECO are also compared in Table 3.

2.3 Uncertainties

Like any other probabilistic analyses, Limerick's PRA contains large uncertainties. The staff is inclined to address this important subject since there is a tendency to focus on the numerical estimates without considering the uncertainties. Uncertainties can be grouped into four general areas: statistical, modeling (assumptions such as human error, common cause models, and others), omissions, and computational. Each of these types of uncertainties is applicable to the various PRA segments such as the core melt sequence estimates, the containment analysis, the source term, and the site/consequence analysis. An excellent discussion on the subject of uncertainty which is also pertinent to this review can be found in "Evaluation of Risk Estimates," in Section II.A of the Indian Point ASLB "Recommendations to the Commission" (ASLB, October 24, 1983). The Indian Point discussion points to two major omissions (sabotage and equipment aging) in the Indian Point PRA which may cause the risk estimates to be low. These omissions apply to Limerick as well. BNL's review of the Limerick PRA and SARA also points out the tendency for modeling assumptions to be conservative in some areas and optimistic in other areas. One could argue that the optimism and pessimism may or may not offset the major omissions. In effect, the overall significance is not known. Given a set of systems modeling assumptions, these uncertainties should be interpreted as being introduced by



1/ Review indicates that the PECO estimate of range of uncertainty seems reasonable.
2/ Estimate is obtained from Table 5.31 of NUREG/CR-3028.

Figure 1 Estimates of the range of uncertainty on core damage frequency

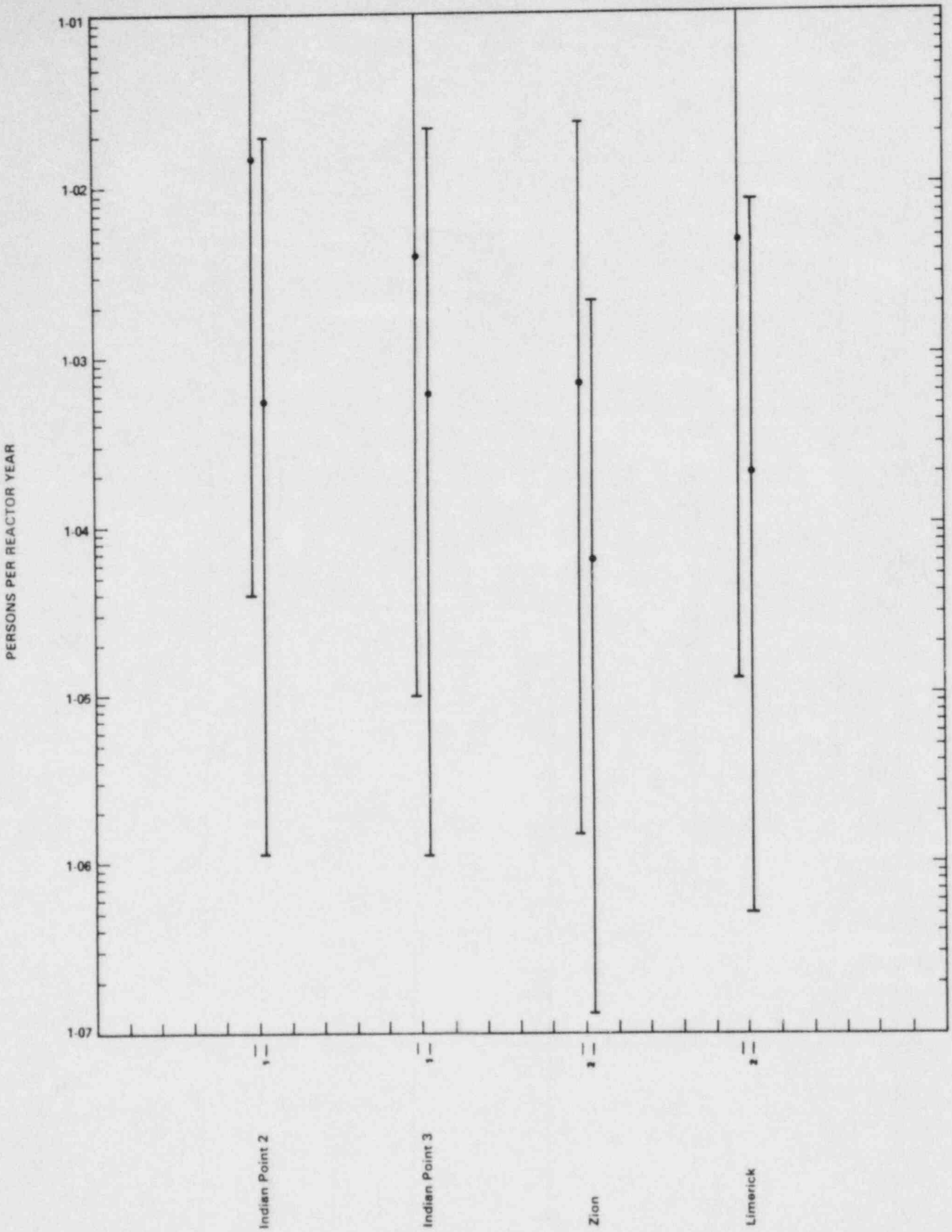


Figure 2 Comparison of Limerick risk with high population density plants - early fatality. (NOTE: Estimates are obtained from staff review, are based on supportive medical treatment and are for the entire region of the plant sites. 1/ Estimate excludes severe seismic and hurricane events. 2/ Estimate excludes severe seismic events.)

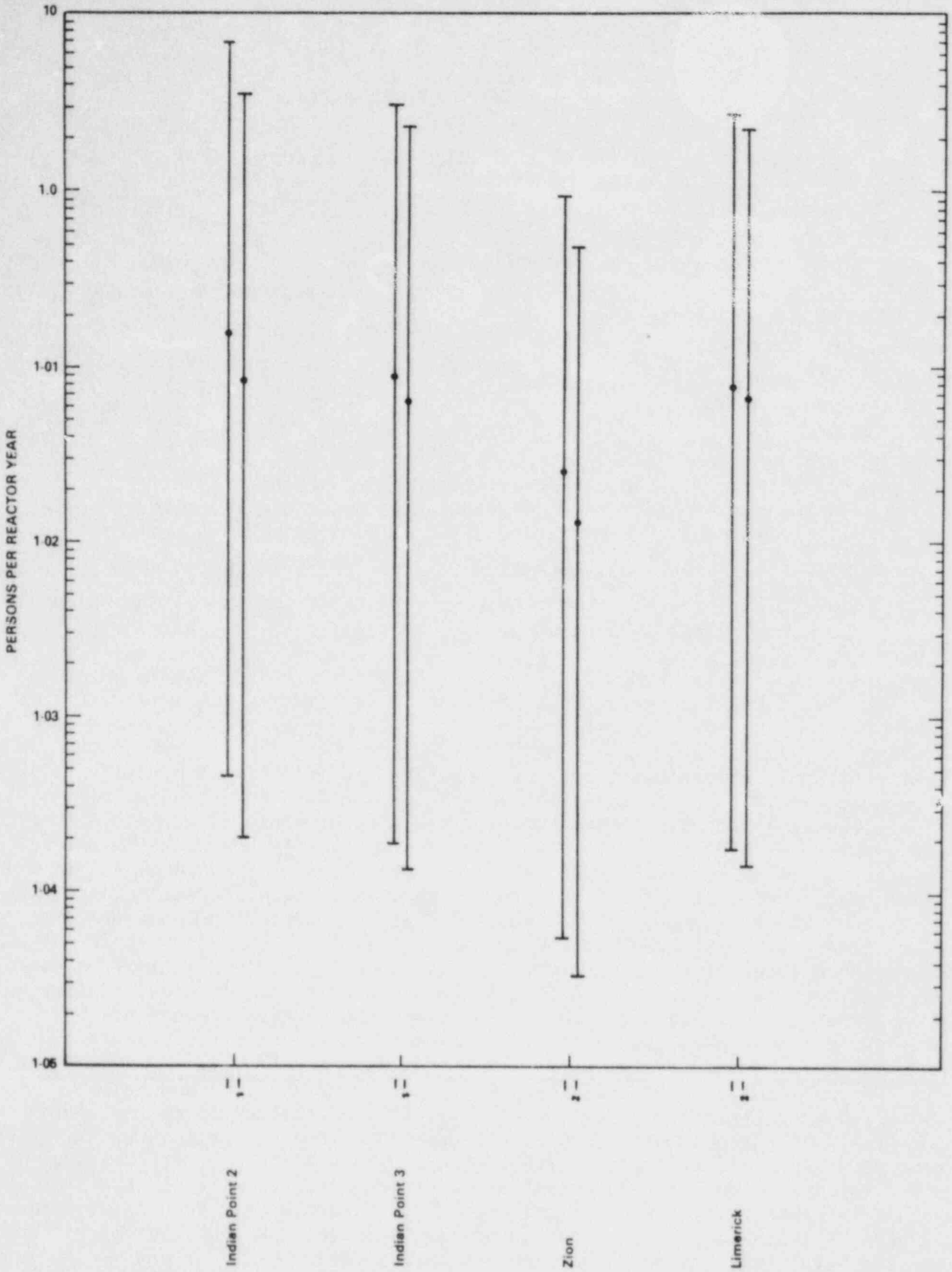


Figure 3 Comparison of Limerick risk with high population density plants - latent fatality (NOTE: Estimates are obtained from staff review, exclude thyroid cancers and are for the entire region of the plant sites. 1/ Estimate excludes severe seismic and hurricane events. 2/ Estimate excludes severe seismic events.)

Table 3 Risk Review of Limerick

Risk Index	PECO1/6/	Review2/6/	Comment
Early fatalities (per plant year of operation)	3.3E-4	5.0E-3	3/, 4/
Latent cancer fatalities (per plant year of operation)	2.8E-2	5.0E-2	4/, 5/,
Person-rems (per plant year of operation)	295	700	

1/Estimates are obtained from Limerick SARA

2/Estimates are obtained from Limerick FES (Table L.1a). See the FES for the uncertainties associated with these estimates.

3/Estimates are based on supportive medical treatment.

4/Estimate are based on crediting those plant modifications which are dicussed in Section 5.

5/Estimates include thyroid cancers.

6/Estimates correspond to "population to 50 miles" case.

uncertainties in the values of various input parameters such as hardware failure data, human error data, frequency of accident initiators (especially loss of offsite power), large and medium LOCAs, fires and seismic events. These sources of uncertainties also exist in the Reactor Safety Study. Some of these sources of uncertainties are still dominant sources because of (1) no data on large LOCAs, (2) relatively sparse data base on severe earthquakes in the eastern United States, and (3) inadequacies in quantifying certain human errors during accident scenarios. There is also large uncertainty attributed to varying degrees of systems success/ failure modeling assumptions, completeness, statistical and arithmetic errors. The impact on the frequency of core damage from this kind of uncertainty is believed to be within range of uncertainties that have been quantified for Limerick.

Both the applicant and BNL have performed sensitivity analyses and have identified potential sources of uncertainty and treated them using the applicable state-of-the-art methodologies. Figure 1 shows a perspective on the range of uncertainty associated with the core damage frequency, as estimated by the applicant and BNL. Note that BNL's estimate of the fifth to the 95th percentile probability range for the frequency of core damage from transients and LOCAs spans almost 2 orders of magnitude (7E-6 to 3E-4 per reactor year) and is greater than the corresponding estimate by PECO.

To provide a perspective on the range of uncertainties on various contributors to the core damage frequency, the staff has provided Table 4 which gives estimates of uncertainty and the mean values of the class of accident sequences contributing to core damage at Limerick. It is evident that seismic events dominate to a large degree the range of uncertainty in each class of accident sequences and, particularly, to the rupture sequence for the reactor pressure vessel which is one of the sequences contributing to the acute fatality risk at Limerick. The high degree of uncertainty specifically associated with the seismic hazard calculations was further emphasized as a result of a recent study by LLNL as discussed in Appendix D.

Although these ranges of uncertainties are very large, the staff believes that the estimates provided in Table 4 are more representative of Limerick than the RSS values. Therefore, the staff used Limerick's specific frequency estimates along with the range of uncertainties in the Limerick Final Environmental Statement (See FES 5.9.4.5(7) for further discussion). The staff believes that the range of uncertainties along with the mean value is very much preferred to the use of a single-point estimate. A more extended discussion regarding the appropriateness of the use of earthquake-related frequencies can be found in Section 4 and in Appendices C and D.

In our discussions of comparative risks and dominant sequences which follow, we summarize the numerical best estimate results of our review. One should use caution in drawing conclusions from these numerical estimates since the uncertainties associated with their derivation (models, data, etc.) and the uncertainties associated with incompleteness are large. The more important use of this report, we believe, should be from the insights gained and the value of potential improvements considering these uncertainties.

These observations about the strengths and weaknesses of PRAs have led the staff to propose a guideline for the reliable use of PRAs during the ASLB hearing on the risk posed by Indian Point: each inference from a PRA to be considered for use in regulatory decisionmaking should be regarded as an hypothesis to be tested, rather than as something to be believed or disbelieved based upon the pedigree of the PRA. One should attempt to identify the assumptions to which the inference is sensitive, and weigh the evidence behind each of these critical assumptions. Alternatively one might catalogue the evidence against the contrary hypothesis. In any case, an inquiry into the relevant sources of uncertainty should be made afresh for each PRA inference, in the immediate context of the inference and its proposed role in safety decisionmaking.

Table 4 Uncertainty Estimates on Various Classes on Accident Sequences at Limerick 1/

Sequence Class2/	Contributor3/	Frequency Range/RY Low	Median	High	Mean Value7/
Class I	I	4.7E-6	3.3E-5	3.3E-4	7.7E-5
	S	1.3E-9	1.7E-7	1.7E-5	3.2E-6
	F	1.7E-7	1.4E-6	1.2E-5	3.4E-6
Class II	I	4.5E-7	2.3E-6	1.1E-5	4.1E-6
	S	4/	4/	4/	5.0E-8
	F	5/	5/	5/	5/
Class III	I	2.6E-7	1.6E-6	1.1E-5	3.3E-6
	S	2.7E-12	1.8E-8	4.9E-6	9.2E-7
	F	5/	5/	5/	5/
Class IV	I	1.7E-8	1.1E-7	1.1E-6	3.2E-7
	S	2.9E-13	2.1E-9	6.7E-7	1.3E-7
	F	5/	5/	5/	5/
Class IS	I	6/	6/	6/	6/
	S	8.0E-14	7.6E-9	7.0E-6	1.2E-6
	F	6/	6/	6/	6/
Class S	I	1.0E-9	1.0E-8	1.0E-7	2.7E-8
	S	1.9E-21	3.2E-11	2.5E-6	4.1E-7
	F	6/	6/	6/	6/

1/For source of uncertainty, refer to Section 2.3 and to Appendix C.

2/For sequence class description refer to Section 2.3.

3/I - Internal (see NUREG/CR-3028, Table 5.31 and SARA Supplement 2, Table 5);
S - Seismic events (see SARA Supplement 2, Table 5);
F - Fires (see SARA Supplement 2, Tables 4 and 5).

4/Estimate is not available.

5/Review indicates that fire does not contribute to these classes of sequences.

6/Not applicable.

7/SARA Supplement 2 Tables 4 and 5 are point estimate values.

3.0 DOMINANT SEQUENCES AND REGULATORY COMPLIANCE

As indicated in Section 2, there are about 86 potential core damage sequences, and these sequences have been ranked with respect to estimates of their frequency. Of these 86 sequences, the five of largest frequency have been reexamined carefully to consider operator recovery actions and evaluate existing design features to mitigate the transients, especially the loss of AC power. These five dominant sequences, major contributors to these sequences, and the applicant's compliance with single failure criteria are described below

3.1 Loss of Feedwater Sequences T_FQUX, T_TQUX, T_EUX, T_IUX

These four sequences are initiated by four groups of transients such as main steam isolation valve (MSIV) closure (T_F), turbine trip (T_T), loss of offsite AC power (T_E), and inadvertently open relief valve (T_I). All these sequences involve failure of the feedwater system (Q) followed by the failure of the high pressure coolant systems (U), such as HPCI and RCIC, to provide coolant makeup to the vessel at high pressure, and the failure of the operator to manually depressurize the reactor vessel to allow the low pressure coolant systems to operate. (These review results were based on an ADS actuation logic that required both high drywell pressure and low vessel water level. As a result of implementation of responses to II.K.3.18 high drywell pressure is no longer an input to the actuation logic.) Since HPCI and RCIC systems are the only systems capable of replenishing lost coolant at high pressure conditions initiated by the above transients, the only remaining way to inject coolant is to lower the pressure in the vessel by using the automatic depressurization system (ADS). The ADS was not previously initiated automatically during the transients because the ADS initiation logic required the high drywell pressure permissive signal which would not be present during the above transients. Thus, the failure of the operator to manually initiate the ADS and reduce the pressure would have foreclosed the choice of using the low pressure coolant systems such as low pressure coolant injection (LPCI) and low pressure core spray (LPCS). This scenario would have resulted in the vessel water level eventually (~25 minutes) decreasing to the top of the active fuel region and damage to the core as it heats up.

The frequency of these four sequences was estimated to be about 6E-5 per reactor year. They contribute about 61% to the total core damage frequency. The dominant contributors to these sequences are: (1) operator failure to manually depressurize the vessel, (2) HPCI and RCIC turbine failures to start, and/or (3) HPCI and RCIC test and maintenance unavailability and/or, (4) failure of the high pressure steam motor-operated valve. However the applicant's ADS logic changes in response to II.K.3.18 is expected to result in a significant reduction in the frequency of the above 4 sequences.

3.2 Station Blackout Sequence T_EUV

This sequence is initiated by the loss of offsite AC power (T_E) followed by the common mode failure of all four diesel generators disabling the high pressure (U) and low pressure coolant systems (V) to provide coolant makeup to the vessel. The sequence also consists of no timely (~4 hours) recovery of offsite AC power and no timely repair of diesel generators. Although high pressure coolant systems such as HPCI and RCIC are independent of AC power, the batteries needed for the start and control of HPCI and RCIC systems will likely be discharged after about 4 hours unless either onsite or offsite AC power is recovered. Thus, according to the PRA, if AC power is not recovered within 4 hours after a station blackout scenario, all the core cooling systems such as HPCI, RCIC, LPCI, and LPCS could stop functioning, and the core damage will commence.

The frequency of the T_EUV sequence is estimated to be about 2.0E-5 per reactor year and it contributes about 22% to the total core damage frequency. The major contributors to the sequence are: (1) common mode failure of all four diesel generators, (2) failure to recover offsite AC power within 4 hours, and (3) failure to provide alternate methods of cooling HPCI and RCIC rooms within 2 hours after the loss of Division 1 and 3 diesel generators.

3.3 Compliance With the Regulatory Single Failure Criterion

The staff evaluated the dominant internal event sequences to check whether these sequences are attributable to noncompliance with the regulatory single failure criterion. The identification of the single failures of these dominant sequences was basically done by linking logically the fault trees of the front line core cooling systems and fault trees of the associated support system, reducing the functional fault trees by use of appropriate Boolean laws, generating the minimum cutsets and evaluating these cutsets to identify single failures, if any. To evaluate all five dominant sequences, BNL examined a total of six cases. They are: a) Low Pressure Coolant Injection (LPCI), b) Low Pressure Core Spray (LPCS), c) High Pressure Coolant Injection and Automatic Depressurization System (HPCI and ADS), d) ADS and Reactor Core Isolation Cooling (RCIC), e) ADS and LPCI, and f) ADS and LPCS. The study was based on the Limerick system fault trees, as revised by BNL, and minimal cutsets were generated. In addition to single failures, doubles were also investigated qualitatively to determine if they can potentially become single failures.

In summary, some individual failures contribute to the dominant accident sequences, i.e., common mode miscalibration of level sensors, diesel common mode failures, DC power supply common mode failures, condensate storage tank failure (CST), and suppression pool failures. However these failures are not considered as "single failures" under current regulatory requirements. A review of results, reported in NUREG/CR-3028, indicates that the dominant sequences are not attributable to unique design features.

4.0 EXTERNAL EVENTS

As stated in Chapter 12 of the Limerick Severe Accident Risk Assessment (SARA) "This study integrates the results of the LGS PRA, which considered accidents occuring as a result of internal initiating events, with the consequences of accidents resulting from external initiating events. The external events that were considered are the following: (1) Earthquakes, (2) Fires, (3) Floods (internal and external to the plant), (4) Tornadoes, (5) Transportation Accidents, and (6) Turbine Missiles. Only earthquakes and fires were found to contribute significantly (i.e., more than a few percent) to either core melt frequency or risk, and therefore only these are discussed explicitly in this chapter."

The NRC staff also believes that it is apparent that the principal external contributors are the seismic and fire hazards. The evaluation of the seismic hazard is discussed in Section 4.1 below and in Appendices C and D. The fire hazard is discussed in Section 5.4 and is also evaluated below. While it is apparent that seismic and fire are the principal contributors for the Limerick site the staff also considered the other external hazards as discussed below.

4.1 Seismic Events

As also discussed in Appendix C, there are several fundamental shortcomings of the seismic event PRA which are inherent in the problem itself and it is beyond our means to adequately address them. First and foremost is the inadequacy of the existing historical and instrumental seismic record (two to three hundred years). PRAs try to utilize this record to draw inferences on earthquakes that appear to have mean return periods from two to four orders of magnitude or more beyond the record. This extrapolation of numerical estimates is speculative, particularly since we lack a fundamental understanding of the causative mechanism of earthquakes in the Eastern U.S. Attempts to deal with the problem lead to the observation that most of the calculated uncertainty in seismic event PRAs, such as LGS-SARA, is related to uncertainty in the seismic hazard. A second problem relates to the fact that the characterization of fragility is based on little data and a great deal of judgement. Finally there are some aspects of the problem where useful and comprehensive models incorporating judgement have not even been proposed. Design and construction errors fall into this category. As a result, there exists a significant potential for systematic bias that cannot be simply accounted for. However when making relative comparisons, that is ratios, where such biases may be common to the entities being compared (e.g. determining which are the major contributors to seismic risk), then errors resulting from them tend to be minimized. We therefore agree with our consultant (BNL) who states that "the results from the LGS-SARA are useful in a relative sense and should not be viewed as absolute numbers."

Reliance upon simple point estimate such as means or medians to characterize actual risk may be premature. However there has been an extensive effort to define the uncertainty. The wide bands of uncertainty presented in relation to the seismic elements of the LGS-SARA can be thought as representing a large part, but not all, of the actual uncertainties. They may be used to gain insight as to the range of the actual risk associated with seismic initiating events at Limerick. We do not mean to imply that higher risk elements (e.g. 95th percentile) are more appropriate than the median, mean or lower (5th percentile) estimates. Indeed the most significant earthquake damage anywhere within the vicinity of the Limerick Site, in the two to three hundred years during which we have records, are fallen chimneys 50 kilometers away during an earthquake at Wilmington, Delaware in 1871 whose magnitude can be estimated to have been less than 5.0. We certainly cannot exclude from the range of reasonable assumptions the judgement that there essentially is no risk to the public resulting from earthquake-induced damage at the seismically-engineered nuclear power plant at Limerick during its operating life.

The nature of seismic PRAs such as the LGS-SARA requires us to look at the behavior and fragility of plants at ground motion levels well beyond the SSE as evidenced by Table 3.1 (Significant Earthquake Induced Failures) of the LGS-SARA. Even though some of these ground motion levels may appear extremely high for such a seismically quite site as Limerick, they provide insight on the seismic capacity of the plant. For example, the applicant in response to NRC questions, estimates that the reactor and control buildings shear walls have a 95% confidence of less than a 5% failure fracture at approximately twice the SSE. Although such conclusions are based upon the generalized assumptions needed to carry out the LGS-SARA, it is our judgement based on past experience that a detailed seismic margins analysis would support the conclusion that the Limerick Generating Station can withstand postulated earthquake ground motion well beyond that defined by the SSE.

Finally a seismic PRA affords an opportunity to examine postulated accident chains and sequences that could lead to serious damage and result in radioactive release. Our review of the LGS-SARA indicates that there are no meaningful outliers in Table 3.1 of the SARA such that simple modification to any of these structures, components or equipment would result in a significant reduction in risk to the public.

Our review and that of BNL have indicated areas of the earthquake related portion of LGS-SARA that could potentially be improved by additional clarification and sensitivity studies. These recommendations, outlined in the sections of Appendix C and NUREG/CR-3493 on seismic hazard and structural, mechanical, component and equipment fragilities, address a wide range of specific seismological and engineering topics. While, potentially, the implementation of these recommendations could result in improvement in some of the areas mentioned it is the staff's position that these additional studies would have no qualitative effect on conclusions regarding the design or safe operation of the plant. Accordingly while the information provided in Appendix C and in NUREG/CR-3493 is useful in support of the remainder of the staff's Risk Evaluation Report the recommendations have not been adopted by the staff for further action by either the staff or the applicant.

4.2 Fire Events

The Severe Accident Risk Assessment (SARA) for the Limerick Generating Station (LGS) has analyzed the accident sequences resulting from fires. This analysis has been reviewed by Brookhaven National Laboratory with the results of the evaluation reported in NUREG/CR-3493 "A Review of the Limerick Generating Station Severe Accident Risk Assessment." By Supplements to the SARA dated July 15, 1983, and November 17, 1983, the applicant has provided an assessment of the impact of plant design changes made since Revision 1 of the Limerick Fire Protection Evaluation Report (1981) which was the version on which the SARA was initially based.

We have reviewed these analyses to check whether the fire risk dominant sequences are attributable to failure to satisfy NRC fire protection requirements or are attributable to a unique design aspect.

The methodology used is consistent with the fire growth modeling and probabilistic risk assessment techniques we have seen employed by others and has two primary shortcomings:

(1) The limited data base on nuclear power plant fires does not permit precise probabilities to be assigned to fire events; and

(2) To date, no fire growth model has been shown to have the capability to accurately predict the growth of fire by correlation of the model to full scale fire tests.

The LGS dominant fire risk sequences identified in the initial SARA submittal were attributable only to non-compliance with our guidelines. That is, the areas identified as high fire risk contributors were the same areas identified as having deficient fire protection when we made our deterministic review of the plant fire protection against our guidelines. The initial SARA did not assess the current plant configuration. In the previous design, redundant safe shutdown equipment in areas identified as dominant fire risk contributors were either not protected or were protected by 30-minute fire barriers. In the present design, these redundant cables and equipment are protected by 3-hour rated fire barriers to comply with our guidelines.

The applicant has recently reassessed the impact of these design changes and found that they have reduced the fire risk contribution, and has documented the reassessment in Supplements 1 and 2 to the SARA.

The staff concludes that further analysis of the SARA concerning fire risk would not be beneficial. The risk of fire has been shown to be within acceptable limits and is not attributable to a unique design concept.

4.3 Flooding Events

The staff reviewed the portion of the SARA which deals with flooding (Chapter 5). The staff's review indicates that the analysis used a logical and systematic approach, and was thorough, comprehensive, and conservative.

The results of the analysis is that the frequency of Class I and Class II accidents resulting from internal flooding is less than 5×10^{-7} and 7×10^{-8} per year, respectively. The analysis addressed internal flooding due to external events (floods, rain, tank failures) and due to internal failures (pipes, pumps, tanks). This analysis is similar to the information in the FSAR in that credit is given for the nonseismic Category I equipment and floor drainage systems. It is more conservative than the FSAR in that a compartment which is being flooded from an internal flooding event is assumed to result in the total loss of everything within the compartment, and less conservative by giving credit for non-watertight doors and seams. The probability of coincident random equipment failures was also included in the analysis. While this analysis method would not be acceptable in accordance with the Standard Review Plan for deterministic safety reviews, it does represent what one would reasonably expect to occur for the given flooding events.

Thus, we find the flood-related portion of SARA acceptable.

4.4 Tornadic Events

The Limerick SARA includes a model for the calculation of the effects of tornado missiles on total plant risk. The staff reviewed the assumptions made by the applicant in this model as to the physical layout, missile availability and behavior of the plant systems under tornado conditions and finds them to be reasonable.

The staff has not reviewed the meteorological assumptions or the probabilistic analysis associated with the tornado missile risk calculation for Limerick. The deterministic safety review of the Limerick plant against the Standard Review Plan has shown that the plant is tornado missile protected except for the ultimate heat sink. The deterministic safety review of tornado missiles indicates that the Limerick ultimate heat sink (spray pond piping structures and spray nozzles) is not protected against design basis tornadoes. The applicant has submitted analyses as part of the deterministic safety review, to demonstrate that the ultimate heat sink can withstand tornadoes consistent with the criteria in the Standard Review Plan. Since this involves showing that the probability of significant damage to structures, systems and components required to prevent a release of radioactivity in excess of 10 CFR Part 100 following a missile strike, assuming loss of offsite power, shall be less than or equal to a median value of 10^{-7} per year or a mean value of 10^{-6} per year, the staff concludes that this will also be a sufficient response to tornado missile related severe accident concerns for the Limerick plant.

The results of the deterministic safety review discussed above will be reported in a supplement to the Limerick SER.

4.5 Turbine Missile Events

As also stated in the Limerick Safety Evaluation Report, the probability of unacceptable damage as a result of turbine missile (P_4) is generally expressed as the product of (1) the probability of turbine failure resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing (P_1), (2) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems or components (SSC) (P_2), and (3) the probability of struck SSC failing to perform their safety function (P_3).

According to SRP 2.2.3 (NUREG-0800) and RG 1.115, for the deterministic safety review, the probability of unacceptable damage from turbine missiles should be less than or equal to about one chance in 10 million per year for an individual plant ($P_4 \leq 10^{-7}$ per year).

In the past, analyses for construction permit and operating license reviews assumed the probability of missile generation (P_1) to be approximately 10^{-4} per turbine year based on the historical failure rate. The strike probability (P_2) was estimated (SRP 3.5.1.3) based on postulated missile sizes, shapes, and energies, and on available plant-specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. The damage probability (P_3) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems (P_4), which is the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic places the regulatory emphasis on the strike probability. That is, having established an individual plant safety objective of about 10- per year, or less, for the probability of unacceptable damage to safety-related systems as a result of turbine missiles, this procedure requires that P_2 be less than or equal to 10^{-3}

Although the calculation of strike probability (P_2) is not difficult in principle, for the most part reducing it to a straightforward ballistics analysis presents a problem in practice. The problem stems from the fact that numerous modeling approximations and simplifying assumptions are required to make tractable the incorporation into acceptable models of available data on the (1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with or perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tend to have a large effect on the resulting value of P_2. Similarly, a reasonably accurate specification of the damage probability (P_3) is not a simple matter because of difficulty of defining the missile impact energy required to make given safety-related systems unavailable to perform their safety function, and the difficulty of postulating sequences of events that would follow a missile-producing turbine failure.

Because of the uncertainties involved in calculating P_2, the NRC staff concludes that P_2 analyses are "ball park" or "order of magnitude" type calculations only. Based on simple estimates for a variety of plant layouts, the NRC staff further concludes that the strike and damage probability product can be reasonably taken to fall in a characteristic narrow range that is dependent on the gross features of turbine-generator orientation because (1) for favorably oriented turbine generators, (P_2 P_3 tend to lie in the range 10^{-4} to 10^{-3}, and (2) for unfavorably oriented turbine generators, P_2 P_3 tend to lie in the range 10^{-3} to 10^{-2}. For these reasons (and because of weak data, controversial assumptions, and modeling difficulties), in the evaluation of P_4, the NRC staff gives credit for the product of the strike and damage probabilities of 10^{-3} for a favorably oriented turbine and 10^{-2} for an unfavorably oriented turbine, and does not encourage calculations of them. In the opinion of the NRC staff, these values represent where P_2 P_3 lie, based on caculations done by the NRC staff and others.

It is the view of the NRC staff that the NRC safety objective with regard to turbine missiles is best expressed in terms of criterion applied to the missile generation probability which requires the demonstrated value of turbine missile generation probability (P_1) be less than 10^{-5} for initial startup and that corrective action be taken to return P_1 to this value if it should become greater than 10^{-5} during operation.

It is the staff's view that the probability of unacceptable damage to safety-related structures, systems and components as a result of turbine missiles is acceptably low (i.e., less than 10^{-7} per year) provided that the above criterion on turbine missile generation is met. This criterion is to be met by the maintenance of an appropriate inservice inspection and testing program on the turbine throughout the plants life as discussed in detail in the Limerick SER.

The applicant's SARA presents a traditionally calculated value of 9.7×10^{-2} for P_2 and presents an argument that this may be conservative by a factor of 10.

The applicant's SARA also presents an argument that based on the scabbing characteristics of impacted concrete and the probability of safety-related SSC being located at the impacted points, the traditionally assumed value of 1 for P_3 may be conservative by a factor of 10.

The applicant also presents an argument that damage to equipment should not be equated on a 1-for-1 basis with core melt. The applicant has evaluated the likely impact areas and has determined that redundant shutdown systems near these areas are generally separated from each other and on this basis introduces an additional factor of 10^{-2} on the probability of core melt.

As noted above the staff's deterministic safety review of the turbine missile issue does not depend on the applicant's analyses of the strike and damage probabilities (P_2 P_3), however, the staff does note that the applicant's approach and results of the calculations (i.e., $10^{-3} \lesssim P_2 P_3 \lesssim 10^{-2}$) are consistent with those reported by other applicants with similar turbine orientations. The staff feels that there is a significant degree of uncertainty, perhaps in both the conservative and nonconservative directions, in the applicant's arguments regarding the product of P_2 P_3 and in the conditional probability of core melt given impacting of SSC. Nevertheless, the staff feels that in conjunction with the condition that the turbine inservice inspection program's objective is to ensure a P_1 of 10^{-5} or smaller, there is substantial supporting evidence that the probability of core melt from a turbine missile initiated event is likely to lie in the range of 10^{-7} to 10^{-8} and therefore constitutes but a small percentage of the core melt frequency and risk.

5.0 PLANT IMPROVEMENTS INFLUENCED BY THE PRA

During the conduct of the Limerick PRA, the applicant recognized the fault tree and event tree techniques as powerful tools to identify dominant sequences and to gain many insights into the strengths and weaknesses that affect independent functioning of the various safety systems at the Limerick facility. Thus, using PRA models, the applicant has developed perspectives on the safety profile of the plant. The applicant has also identified improvements and changes in plant design and/or improvements to existing operating procedures that can significantly reduce risk. These additional improvements are beyond those required by the staff's standard review plan reviews which are also applicable for Limerick licensing. Table 5 is a list of major plant modifications that are influenced by the PRA and the resulting safety improvements. These plant modifications are summarized below.

5.1 ATWS Improvements

Many hardware modifications and procedural changes belong to this category. The applicant has increased the reliability of the scram sensing system by installing additional (redundant and diverse) sensors in the scram instrument volume. The standby liquid control (SLC) system has additional flow capacity (3 pumps delivering 129 gpm versus 2 pumps providing 86 gpm) and provides improved reliability to overall scram function. The SLC system has two redundant containment penetrations and, therefore, the system can perform its safety action in the event of a pipe break in one of the penetrating pipes (1.5-inch diameter). The SLC system is automatically initiated in the event of scram failure and, therefore, the system is not dependent on the operator who is not assumed to perform the required function reliably in fewer than 5 minutes following the postulated scram failure. The system can also be tested on line without reducing its availability.

The applicant has also provided alternate rod insertion (ARI) capability which provides further assurance that reactor trip will occur in the event of electrical failure of the reactor protection system.

During the study of ATWS risk scenarios, the applicant recognized the difficulties in removing heat while keeping the reactor at high pressure. Because the MSIV closes at vessel water level L2, the feedwater (FW) system providing high pressure coolant is lost because of its dependency on main steam. Therefore, the applicant modified MSIV closure logic so that the MSIVs now close at vessel water level L1.* Because it takes time for the water level to decrease from L2 to L1, heat can be removed through the normally operating feedwater (FW) system and open MSIVs for a longer time. Thus, the MSIV trip logic change from L2 to L1 improves the availability of the FW system in removing heat not only during ATWS scenarios, but also during other transients. Also,

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*L1 is about 1.5 feet above the top of the fuel.
L2 is about 10 feet above the top of the fuel.

Table 5 Voluntary Plant Improvements Influenced by the PRA

Items	System Reliability Improvement Factor 3/ 5/	Sequence Frequency Reduction Factor 4/ 5/
ATWS Alternate 3A fixes. These include SLC pumps (129 gpm) improved automatic system initiation, on-line test capability, alternate rod insertion, recirculation pump trip, redundant and diverse scram volume instrument sensors and MSIV isolation setpoint change L2 to L1	20	10
ADS air supply system improvements (added redundant air solenoids, piping, and valves)	12	1.2
RHR SW pump discharge crossover valves added	11	11
Containment overpressure relief system	1/	1/
Added fire barriers for reactor building equipment hatches	7	7
Added procedure to reset selected electrical equipments after seismic events	2/	2/
MSIV air supply system improvements	2/	2/

1/ Previously considered and now system is removed.

2/ Estimate is not available.

3/ Reliability improvement factor is the ratio of the reliability estimate before the system modification to the reliability estimate after the system modification.

4/ Frequency reduction factor is the ratio of the sequence frequency estimate before the system modification to the sequence frequency estimate after the system modification.

5/ Estimates were provided by the applicant.

the MSIV trip logic change from L2 to L1 reduces the frequency of vessel isolation transients. HPCI and RCIC will still initiate at level L2 and MSIVs will close on high steamline radiation. Therefore, the MSIV trip logic change from L2 to L1 does not involve a significant reduction in the ability to contain fission products. These ATWS improvements are expected to decrease the frequency of severe ATWS sequences by about a factor of 10 and have, therefore, significantly reduced the frequency of total core damage. Because ATWS-induced sequences (particularly Class IV sequences) were, prior to the ATWS modifications discussed herein, the major contributor to early fatality risk, the early fatality risk at Limerick has been reduced significantly, by at least a factor of 10. The applicant has voluntarily made substantial modifications to the Limerick design which provide protection against ATWS beyond what would be required by the soon to be published new rule on ATWS.

5.2 Air Supply System Improvements

The automatic depressurization system (ADS) requires an air supply system, as one of the support systems. When the applicant quantified the hardware reliability of the ADS, failure of the air support system was found to be the major contributor to the ADS unreliability. Further investigation of the air support system's fault tree revealed a few weak spots such as single train air piping, single pilot solenoid valve, and undesirable location of the gas supply system. The applicant has corrected these vulnerabilities in the air supply system by installing dual pilot solenoid valves and redundant air piping. The type and location of the backup gas supplies have also been redesigned to improve the safety-grade to non-safety-grade interfaces. The system has been made substantially more reliable.

5.3 RHR and Containment Venting Improvements

The residual heat removal (RHR) system requires the RHR service water system as one of the support systems. When the applicant quantified the containment heat removal (TW) type sequences, the importance of RHR service water system unreliability for long-term heat-removal purposes and the adverse system interactions resulting from the loss of the RHR service water support system were recognized. The loss of the RHR service water system following a transient will totally disable the RHR system which, in turn, will result in heatup and overpressurization of the suppression pool. Thus, if the pool pressure is not sufficiently reduced either by recovering the lost RHR service water system or by venting the containment, the high pressure coolant systems (HPCI and RCIC) will stop functioning because they are dependent on the shaft-driven lube oil cooling system which takes suction from the hot and pressurized pool. The HPCI lube oil system is assumed to fail at a pool temperature of 200°F. Although low pressure coolant systems are designed to be capable of discharging saturated water for the above scenarios, there was still a concern about the capability of the low pressure coolant system because of its dependency on the ADS.

The crossover valves of the RHR service water system were identified as the potential source for improving reliability. These crossover valves enable the cross connection of Unit 2 RHR service water pumps to the Unit 1 RHR heat exchangers and vice versa. The applicant will have the necessary portions of the Unit 2 RHR service water system operational prior to Unit 1 fuel load. This modification enhances the redundancy of RHR service water supply to RHR heat exchangers of both units. This particular RHR service water system modification has resulted in improving the reliability of the RHR system by a factor greater than 10. Also, the applicant has, with this relatively low-cost modification, increased the heat removal capacity of the containment and has reduced the frequency of sequences involving failure of containment heat removal function (TW) significantly.

The applicant has also committed to establish an emergency procedure and training program to vent the containment during containment overpressure scenarios, using existing equipment, in order to avoid containment overpressure failure and resulting core damage. The existing equipment that could potentially be used for venting the containment at Limerick range from small lines, some of which have some filtering capability, to the large purge lines. This matter has been under active review by the staff as part of the deterministic safety review of the Limerick Procedures Generation Package submitted in response to TMI Action Plan item I.C.1. The staff intends to review the proposed venting criterion (both generically and for each plant) as an item typically considered long term under Item I.C.1.

5.4 Fire-Related Modifications

The applicant's initial assessment of fire risk in SARA was based on Revision 1 of the Limerick Fire Protection Evaluation Report as submitted in 1981.

When the applicant assessed fire risk on this basis, the fire sequences were estimated to be the dominant contributor (about 53%) to the total frequency of core damage (see SARA Table 12-1). The applicant investigated the fire propagation event trees, seeking possible areas of improvement. All safety-related cables for alternate shutdown methods A and B* were protected by only 30-minute or 1 hour fire barriers where spatial separation did not suffice. Therefore, the applicant upgraded the protection for shutdown method B cables to withstand fire for a period of 3 hours where spatial separation did not suffice. This particular modification has not only reduced the probability of cable failure significantly (for up to 3 hours), but also has provided some additional recovery time, in case the fire suppression system does not function initially. Overall, upgraded fixes for fire-related items have resulted in significant improvement in the fire integrity of shutdown method B cable (by a factor of about 30 to 40). In addition the applicant also systematically investigated areas vulnerable to fire, using PRA fire zone models, and upgraded the protection for the equipment hatch in the reactor building (RB) by installing fire protection barriers for the equipment hatch and, therefore, has adequately separated the redundant shutdown systems.

The fire-related modifications described above have made fire sequences small contributors to overall core damage frequency.

*Shutdown methods are as defined in the Limerick Fire Protection Evaluation Report, Revision 4.

6.0 A PERSPECTIVE ON COMPARISON OF LIMERICK WITH OTHER PLANTS

6.1 Comparison of Limerick's Frequency of Core Damage With That at Other Plants

Currently, probabilistic risk analyses have been published for at least 13 U.S. nuclear power plants. These 13 plants (listed in Table 6) represent the full range of designs that have been built in the United States. Four of these 13 plants (Indian Point 2 and 3, Zion and Limerick as shown on Figure 4) are located at the highest population density sites. A significant amount of reliability and risk information has become available which bears on the question of comparison of various nuclear power plants. However, having this comprehensive information available has posed as many new questions as it has helped to answer. Questions concerning expanded scope, consistency of approach, adequacy of data, level of detail, and analytical quality assurance are being raised as potential areas where comparison between studies could be faulted. The state of the art of PRAs is evolving rapidly and continuously. Large amounts of resources in both private and government research are being expended to improve PRA methods and data.

Although Figure 4 indicates that the Limerick core damage frequency is lower than that of other high population density plants, caution must be exercised when using these results since very large uncertainties exist in these analyses. No attempt has yet been made to adjust the results to compensate for inconsistent approaches or methods. Therefore, the appropriateness of the comparison may be questioned.

6.2 Comparison of Limerick's Risk With That of Other Plants Having a High Density of Population

For radii up to 50 miles, the population surrounding Limerick is roughly 10 times that of the median site in the United States (see NRC, August 1980), and the population is roughly half that found around Indian Point Units 2 and 3.

To provide a perspective on how the Limerick facility compares in terms of risk from core damage accidents with the above nuclear facilities located near areas of high population density, the estimated risk from core damage accidents for Limerick and Indian Point Units 2 and 3 is shown in Figures 2 and 3 in Section 2.3. The values shown in these figures include risk from internal and external events and correspond to the staff's review of the above plant-specific PRAs. For each risk index, uncertainties are also indicated in the form of range to provide a proper perspective for comparison on associated uncertainty. Although these figures indicate that the societal risks posed by core damage accident at Limerick are about the same as the risk posed by core damage accidents at Indian Point Units 2 and 3, some caution should be observed when interpreting the results. This is because different methods and data were used in these PRAs. This should be recognized when comparing relative risk estimates until data and methods used on various PRAs have been rebaselined.

Table 6 U.S. Nuclear Power Plants That Have PRA Analyses

Plant	PRA Sponsor	Document	Yr of Pub.	NSSS	AE	Power MW(e)
ANO-1	IREP/NRC	NUREG/CR-2787	81	B&W	Bechtel	836
Big Rock Point*	Consumers Power	BRP PRA	81	GE	Bechtel	71
Calvert Cliffs	RSSMAP/NRC	NUREG/CR-1659	82	CE	Bechtel	850
Crystal River	IREP/NRC	NUREG/CR-2515	80	B&W	Gilbert	825
Grand Gulf	RSSMAP/NRC	NUREG/CR-1659	81	GE	Bechtel	1250
IP-2*	PASNY/CON ED	IPPSS	82	W	UE&C	873
IP-3*	PASNY/CON ED	IPPSS	82	W	UE&C	965
Limerick*	Phil Elec	LGS PRA/SARA	81	GE	Bechtel	1055
Oconee	RSSMAP/NRC	NUREG/CR-1659	80	B&W	Bechtel	886
Peach Bottom	RSS/NRC	WASH-1400	75	GE	Bechtel	1065
Sequoyah	RSSMAP/NRC	NUREG/CR-1659	78	W	TVA	1148
Surry	RSS/NRC	WASH-1400	75	W	S&W	775
Zion*	Commonwealth Edison	ZPPSS	81	W	S&L	1100

*Included a risk assessment incorporating "externally" initiated events.

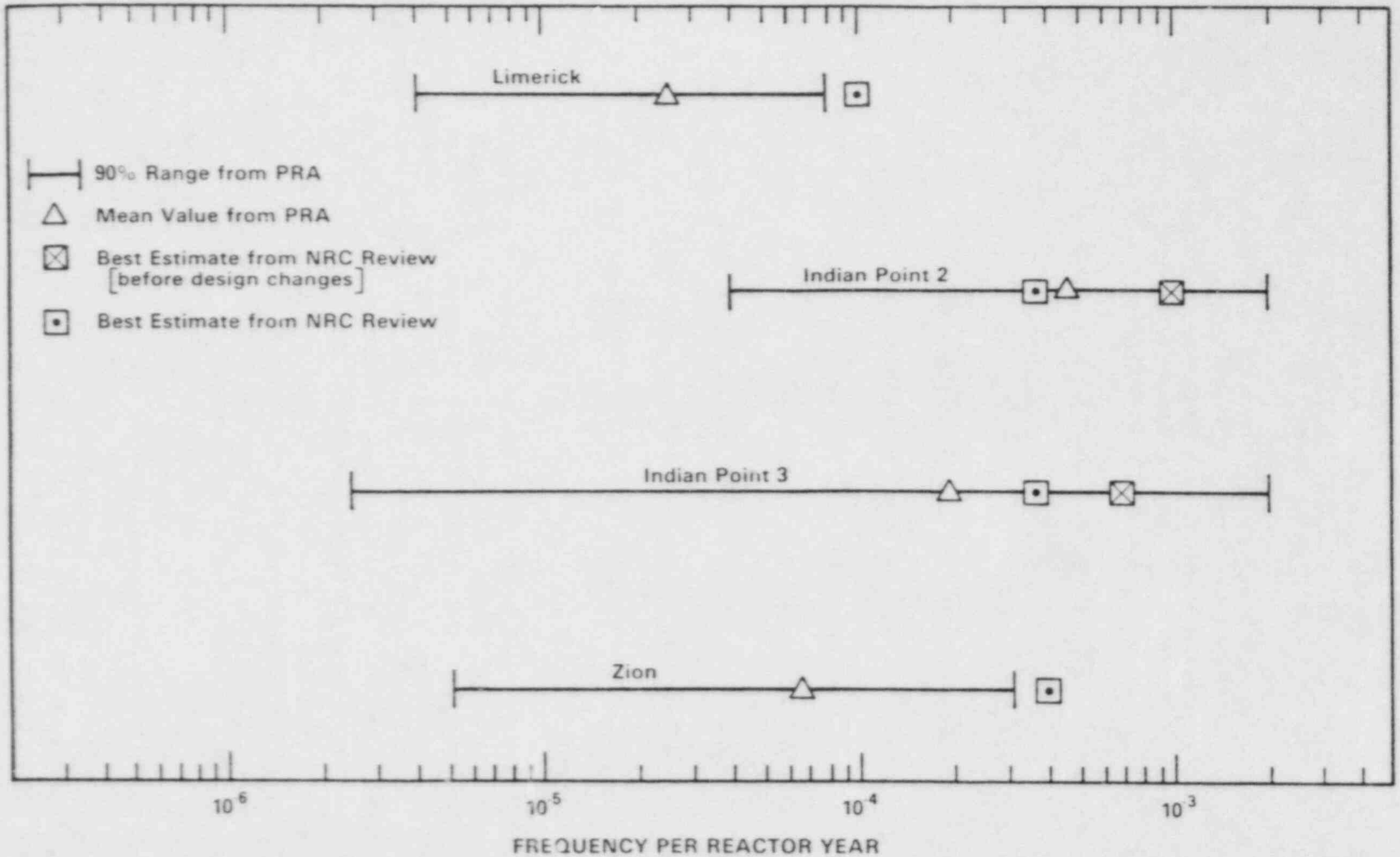

Figure 4 Comparison of Limerick core damage frequency with high density population plants - uncertainty range of internal and external events.

NOTE:

The PRAs were not necesssarily performed using consistent methodologies or assumptions. Many of the PRAs evaluate designs that have subsequestly been altered.

6.3 Comparison of Limerick's Accident Sequences With That of the Reactor Safety Study (RSS) BWR

In addition to the comparison of Limerick with other plants that have high surrounding population densities, the staff provides some perspective on comparison of Limerick with the Reactor Safety Study (RSS) BWR plant.

The staff's comparison of Limerick's core damage frequency with that of the RSS BWR follows. At this time, Limerick's core damage frequency estimates appear to be higher than RSS-BWR core damage frequency estimates by a factor of about 3. However, the applicant has in several respects improved the Limerick design incorporating greater redundancy than does the BWR analyzed in the RSS.

(1) 4 diesels per unit, each with redundant air start systems and redundant emergency service water (ESW) supplies;

(2) 4 separate electrical divisions;

(3) more and better arranged offsite power sources;

(4) redundant cooling loads assigned to separate ESW loops;

(5) RHR pump discharge cross-ties;

(6) design of ESW/SW interfaces;

(7) auxiliary steam supplies to steam jet air ejectors;

(8) flexibility in use of spray pond and cooling towers;

(9) redundant, series suppression pool/drywell vacuum breakers;

(10) establishment of appropriate fire zones.

Also a perspective on the design differences between Limerick and RSS BWR plant is provided in Table 7. These inherent features along with the voluntary improvements influenced by the PRA should have yielded (all other things being equal) lower core damage frequency estimates for Limerick than for the RSS BWR. The higher estimate of Limerick core damage frequency is primarily because of the differences in estimates of the dominant sequences (TQUX, T_EUV). The frequency of TQUX and T_EUV sequences is somewhat underestimated in the RSS plant because simple models (such as one single-transient event tree) were used and because of insufficient human reliability analysis. However, the applicant and BNL have used more realistic methods and updated information to quantify these sequences for Limerick than were used in the RSS. Therefore the frequency of these sequences is estimated to be higher than RSS-BWR estimates. In contrast to Limerick's dominant accident sequences, the RSS-BWR study concluded that sequences involving transients followed by the total loss of the containment heat removal function (TW) are the type of accident with the highest frequency. Anticipated transients without scram (ATWS) sequences and sequences involving transients followed by loss of high and low pressure injection functions (TQUV) occupy second and third rank among the contributors to core damage frequency.

Table 7 Major Design Differences Between Limerick and the RSS BWR

Design Feature	RSS-BWR	Limerick	Effect
Containment design	Mark I (free standing steel)	Mark II (concrete with steel liner)	Different suppression pool configuration, radioactive release pathways and containment failure modes.
Offsite power supplies	2 redundant offsite power supplies	Five off-site power supplies	Decreases the probability loss of power. This results in a lower probability of core melt for certain sequences.
Emergency diesel generator	4, shared between units	4 per unit, no sharing between Units	Increases the reliability of emergency power system.
NPSH requirement on low pressure ECCS pumps	Pumps were failed at saturation conditions	Pumps are designed to pump at saturated conditions	Increases the probability of successful coolant injection under adverse conditions, thereby reducing the probability of a core melt.
ATWS mitigation (Alternate 3A with 3 SLC pumps)	RPT, manual SLC initiation, 2 SLC pumps	a. ARI b. RPT c. Automatic SLC d. FW run-back e. Scram discharge instrument volume modifications	Reduces the probability of an ATWS by prevention (ARI+RPT). Also reduces the probability of an ATWS leading to a core melt by mitigation once it has occurred. Reduces early fatalities significantly.
RHR connections	4 dedicated RHR heat exchangers	2 RHR heat exchangers with the ability to cross-connect to 2nd RHR pump and the Unit 2 RHRSW	An increase in RHR reliability.
Emergency service water (ESW) system	One 100% loop per unit	2 100% loops per unit, shared between units	Increases the reliability of ESW support system.

This inversion in the order of importance is due partly to differences in the methodology used, as well as to the differences in the design of the plants. The methodology employed in the Limerick PRA is more detailed than that employed in the RSS, particularly in its inclusion of the modeling of recovery of unavailable systems such as MSIV re-opening, FW repair and recovery, diesel repair, and recovery of offsite AC power. The system for containment heat removal at Limerick is more reliable than that considered for the RSS-BWR plant. Finally, the ATWS sequences do not contribute significantly to the frequency of core damage at Limerick, because the applicant has incorporated the ATWS Alternate-3A modification as part of Limerick's PRA review. Overall, the applicant has improved the design at Limerick beyond the design of the RSS BWR.

7.0 MARK II CONTAINMENT ADEQUACY

To assist in the staff review, BNL performed independent calculations, using the MARCH/CORRAL system of computer codes to review the applicant's containment response analysis. Using RSS methodology, BNL has reestimated appropriate source terms for each of the 27 release categories shown in the Limerick FES, and has reported in BNL-33835 such results as release fractions of various fission products from the damaged core, height of release, and energy release. The staff has reviewed the containment response analysis reported in both the PRA and BNL-33835 and has obtained some important insights into the strengths and weaknesses of the containment systems and structures of the Limerick Mark II containment. These insights follow.

7.1 Early Failure vs Late Failure

If only internal events are considered, the above assessments indicate that Class I transients have the highest frequency of occurrence at Limerick. It might, therefore, be expected that these sequences would be major contributors to latent fatality risk. However, an important assumption regarding the analysis of Class I sequences relates to the deposition of the core debris as it is released from the reactor vessel. In the Limerick PRA, it was assumed that most of the core debris would be retained on the diaphragm floor in the drywell. Detailed analysis at BNL (NUREG/CR-3028) supports this assumption for the Limerick configuration. This is an important conclusion and implies that containment failure will occur via overpressurization several hours after vessel failure as a result of noncondensible gas generation during interactions between the core and concrete. This delay in containment failure is important because it provides time for attenuation of fission products even assuming WASH-1400 source term methodology. In fact, because of the significant reduction in source terms, both the Limerick PRA and the BNL study indicate that the Class I sequences contribute negligibly to the total early fatality index. Class IV and seismic events dominate the predicted early fatality index. Class IV sequences comprise ATWS events with a core meltdown into a failed containment. However, because of the higher frequency of Class I sequences, the long-term damage indices (latent fatalities and thyroid cancers) are dominated by this class of accidents. It is noteworthy that BNL's best-estimate calculations indicate that as the core materials are released from the reactor pressure vessel (RPV), they will be retained on the diaphragm floor in the drywell. Containment failure occurs many hours after RPV failure from gradual overpressurization during core/concrete interactions or diaphragm floor failure. However, if significant quantities of core debris pass through the diaphragm floor shortly after vessel failure and enter the suppression pool, the subsequent debris/water interactions could have potentially important implications. If the interactions result in containment failure, the fission product releases could be higher than calculated in BNL's best-estimate case. However, if the containment does not fail during the initial debris/water interactions and fails on a very long time scale (or never fails), the fission product releases will be significantly lower than what was calculated in BNL's best-estimate case.

Because of the uncertainty associated with the behavior of ex-vessel core debris, BNL attempted to establish an uncertainty band on consequences by determining corresponding release categories. The upper bound was determined by assuming that most of the core debris falls into the suppression pool after the vessel fails, and that the resulting rapid steaming fails containment. In addition, BNL assumed that most of the fission products associated with the melt release are discharged from the primary system after RPV failure and shortly before containment failure. These assumptions correspond to the early containment failure case. This case applies primarily to Class I sequences and increases the mean latent fatalities, and predicts that early fatalities will occur (note that, in the point estimate calculations, this category was calculated to have zero early fatalities). The upper bound was primarily obtained by increasing the risk associated with Class I sequences as described above. The remaining release categories were not changed. BNL considers this upper bound calculation to be extremely conservative.

The upper and lower bounds indicate that the BNL point estimate calculations are significantly closer to the upper bound than to the lower bound for latent fatalities. Finally, BNL feels that the range of uncertainty encompassed by these limits will include current NRC efforts at being more realistic in determining the release of fission products.

7.2 Drywell Atmosphere Temperature and Impact on Penetration Seals

Coupled with the uncertainty in pressurization of the drywell and wetwell following a RPV failure, there is considerable uncertainty in the drywell airspace temperature. The peak temperature can vary from approximately 300°F to 600°F, depending on the assumptions made in its determination. Parameters which affect this temperature are the interfacial heat transfer coefficients between the steel liner and the concrete containment wall, and the area over which the molten core spreads on the diaphragm floor. The primary concern in this case is failure of containment penetration seals as a result of high temperature. This failure would lead to a bypassing of containment and possible venting of fission products to the environment. A program addressing the problem of leaking seals in containment buildings at elevated temperatures and pressures is currently under way at BNL. Methods developed in this program will be used to investigate whether significant quantities of fission products will leak to the environment as a result of temperature-induced and pressure-induced seal failure. This program is included in the NRC's generic severe accident decisionmaking activities.

7.3 Potential Effect of Existing Systems

A number of existing systems have been found to be important for mitigating severe accidents. Although these systems were installed at Limerick for other than severe accident purposes, their availability bears significantly on the staff's perception of the risk at Limerick. Some major highlights are:

(a) The plant is to be operated with an inerted containment. Hence, hydrogen production associated with a severe accident will not result in burns or detonations and hence may not represent a threat to containment integrity. Early containment failure is associated with ATWS events. Other more probable transients are not predicted to cause early containment failure.

(b) The reactor pressure vessel safety relief valves discharge to the suppression pool. The suppression pool is an efficient fission product scrubber.

(c) The standby gas treatment system (SGTS) is believed to be an efficient filter removing fission products from a potentially leaking containment prior to discharge to the atmosphere. The SGTS is sufficiently sized to serve as a filtered vent for half the Class 1 sequences for which AC power is available. The predicted containment failure mode is leakage less than the SGTS capacity from the containment to the auxiliary building.

The above three features, inerted containment, suppression pool scrubbing, and operation of the SGTS, were modeled in the Limerick PRA and were reviewed by the staff and BNL. However, the containment sprays [CS] were not modeled in the PRA and, therefore, an explicit assessment of their effect on accident sequences was not made. Proper operation of the CS system could significantly increase the containment failure time and could reduce the fission product inventory in the containment atmosphere and, therefore, could reduce risk. In response to the staff's inquiry regarding the CS system availability, the applicant responded by letter dated April 11, 1984. The response indicated that the Limerick emergency procedures (TRIP) make use of the sprays to limit pressure in both the drywell and the wetwell and that the availability of the containment spray mode of RHR system operation will be ensured by monthly surveillance testing. The staff believes that this is a reasonable and sufficient response to this concern.

8.0 ENGINEERING AND OPERATIONAL INSIGHTS IMPORTANT TO SAFETY

8.1 Considerations in Arriving at Recommendations

On the basis of the staff and the BNL review of the Limerick PRA the staff has developed the following insights.

The first inquiry on comparison of risk for plants located in areas of high population density finds (Section 6) that both individual and societal risks posed by severe accidents at Limerick are well within the spectrum of risks posed by severe accident at other plants with high population densities, such as Indian Point Units 2 and 3 and Zion. However, no completely adequate basis exists for the interplant comparison of Limerick with all other nuclear power plants licensed to operate in the United States because of the different methods used, incompleteness, and large uncertainties (see Section 6). The staff has not reviewed all nuclear power plant risk assessments performed to date to the same depth as it has the assessments made for Limerick, Indian Point, and Zion. As a result, risk comparisons between these three high-population density sites and other plants cannot be made at this time with a high degree of confidence. Within our understanding of uncertainties, however, the staff concludes that the risks from Limerick are not appreciably different in magnitude than those of Indian Point or Zion.

The second inquiry on risk of various release categories estimated by the staff finds that no single release category dominated the total risk (early and latent fatality). A careful examination of the applicants' voluntary improvements and the risk profile estimated by the staff indicates that no single sequence dominates the risk significantly. The applicant's voluntary improvements have lowered the frequency of severe release categories by a significant amount and, therefore, have significantly reduced early fatality risk.

The third inquiry on the vulnerability of Limerick's Mark II containment for severe release finds that there is no single containment failure mode that dominates the release from severe accidents at Limerick.

The fourth inquiry on all 86 core damage sequences finds that no single sequence exceeded a frequency value of 10^{-4} per reactor year. Only a few factors contribute significantly to the estimated frequency of core damage and they involve:

(1) Human failure to depressurize the reactor in a timely fashion. However, since the PRA was performed, the applicant has committed to modify the ADS actuation logic to be consistent with the resolution of NUREG-0737 Item II.K.3.18. This modification is expected to reduce this factor as a significant contributor.

(2) Human failure to establish an alternate method of HPCI/RCIC pump room cooling.

(3) Failure to recover offsite power within 2 hours of its loss and subsequent depletion of batteries supplying power to HPCI/RCIC system components beyond four hours following a station blackout.

8.2 Insights and Recommendations

In addition to determining the dominant contributors to the dominant sequences, as discussed above in Section 8.1, these contributors have been carefully evaluated to determine the reasons why these are significant to the dominant sequences. Thus, the staff has obtained some important insights about systems safety. The major insights are summarized below.

8.2.1 ADS Design Change and Impact

The existing design of the ADS actuation logic at the time the PRA was conducted required the presence of both high drywell pressure and low vessel water level signals to initiate ADS automatically. This required that, for some transients and accidents that resulted in a loss of all high pressure coolant makeup but did not also result in high drywell pressure, that the operator take action to manually depressurize the reactor to permit low pressure cooling systems to operate. The staff and BNL found that the procedure guidelines on operator actions in a manual depressurization were confusing. This resulted in a high estimated failure rate for the manual depressurization function (X) in the PRA evaluation and as such was of concern to the staff.

Two actions, both of which are within the scope of the staff's deterministic safety review, have taken place since the early phases of the staff's PRA review, which significantly ameliorate the staff's concerns in this regard.

One, as reported in Section 13.5 of the SER (NUREG-0991) and its supplements, the applicant's procedure generation package and thus its procedures are based upon NRC staff approved BWR Owners Group Emergency Procedures Guidelines. The staff also reviews the Limerick specific procedures generation package. To illustrate the value of improving the procedures over those initially reviewed, an estimate of the sequence frequency reduction resulting from the improved procedure is shown in Table 8.

Second, in response to TMI Action Plan Item II.K.3.18, Modification of ADS Logic, the applicant has committed to implement, prior to fuel loading, logic changes which would involve bypassing of the high drywell pressure trip and the addition of a manual inhibit switch. As stated in Section 15.9.4 of the SER, the implementation of this logic modification eliminates the need for manual depressurization of the reactor vessel for certain events such as a stuck open safety relief valve or a steamline break outside the containment with HPCI failure. As also stated in the SER, the staff concluded that this constituted an acceptable response to II.K.3.18 for Limerick.

The staff believes that the two above actions, which are also summarized in the applicant's April 11, 1984 letter, constitute a reasonable and sufficient response to this PRA concern.

8.2.2 HPCI and RCIC Room Cooling

The loss of high pressure coolant systems (HPCI and RCIC) during the loss of AC power could occur because of the dependence of HPCI and RCIC systems on their room coolers (which are lost during the loss of AC power). Therefore, the staff explored whether some external means of providing forced air circulation

Table 8 Staff's Additional Insights To Improve Plant Risks at Limerick

Item	Dominant Sequence					Core Damage Frequency Reduction Factor 1/
	T_EUV	T_FQUX	T_TQUX	T_EUX	T_IUX	
Improved ADS initiation logic (removal of high drywell pressure permissive signal per TMI Action Plan Item II.K.3.18) following the potential loss of high-pressure coolant sources. 4/	2/	6	6	6	6	2.3
Improved design to achieve alternate method of HPCI/RCIC room cooling during the loss of offsite power events. It may include opening HPCI RCIC doorways and turning on some dedicated fans to cool the room. 3/	1.2	2/	2/	1.2	2/	1.2
Items 1 and 2	1.2	6	6	6	6	2.5

1/ Core damage frequency reduction factor is the ratio of the frequency estimate before the system modification to the frequency estimate after the system modification

2/ Negligible or not applicable.

3/ Dedicated fans may need dedicated batteries.

4/ The improvement factors are based upon the staff's judgement and on a presentation before the ACRS by the applicant's consultant on October 13, 1983.

may be necessary if AC power is not recovered within 4 hours. The Limerick PRA assumes a higher failure probability (0.15) to provide an alternate method of room cooling because the alternate method of room cooling involves only opening the stairway doors and room doors for which procedures were not then developed. The staff explored with the applicant whether or not modifications to the existing method of room cooling (for use in the event of loss of AC power), such as establishment of forced air circulation using battery-operated fans, were needed to reduce the probability that the ECCS (HPCI/RCIC) room cooling function would fail. Adequate room cooling would enable HPCI and RCIC systems to continue functioning independently during a loss of offsite power and would, therefore, reduce the dominant sequence (T_EUV) frequency significantly.

An estimate of the risk reduction of providing adequate room cooling beyond 4 hours is investigated and is shown in Table 8.

In its letter of April 11, 1984, the applicant committed to develop specific procedures to ensure adequate room cooling and to verify its previously performed analyses on the adequacy of room cooling. The staff believes that these actions are reasonable and sufficient to address this concern.

The staff performed a preliminary cost/benefit analysis on the insights in Sections 8.2.1 and 8.2.2. The preliminary cost/benefit analysis indicated that the actions to implement the insights are cost effective. Those actions have already been taken by the applicant as indicated in the letter of April 11, 1984.

8.2.3 Safety Assurance Program

Section 2 discussed the sources and ranges of uncertainty estimates around the Limerick core damage frequency. Also, Section 6 discussed the difficulties involved in comparing the frequency of core damage between plants because of the large uncertainty and different methods and data used to estimate the core damage frequency. In view of large uncertainty in risk estimate around Limerick's already high population density site, it appears reasonable and prudent that the applicant establish and implement a safety assurance program with the objectives of assuring that the conduct of operations and future clues to the safety of the plants are and remain consistent with a level of severe accident risk not appreciably greater than that assessed by the staff. The bases for the staff's recommendation on the safety assurance program for Limerick follows.

First, the staff has observed that many of the more important precursors to severe reactor accidents occurring at other plants entailed maintenance error, surveillance error, operator error, or management oversight. Virtually every historical instance in which whole redundant safety systems have been found to be inoperable can be traced, in part, to such errors. A common element in all these occurrences was a failure by operations personnel to fully appreciate the importance to risk of their own actions or of the systems or phenomena entailed. Procedures and operations staff training altered to reflect the insights obtainable from Limerick's PRA and staff analyses of risk could go a long way to decrease the likelihood of such errors.

Second, the Limerick PRA review and other PRA reviews indicate that PRAs tend to credit operator actions different from or extending beyond that in current emergency procedures and/or operator training. (See discussion above in item 2 on HPCI/RCIC room cooling.) Where the Limerick PRA suggests operator response tactics that result in lower risk than adherence to the written procedures, the procedures should be reexamined and improved, if possible. In one instance a procedural change was made during the Limerick PRA review which would increase HPCI/RCIC availability in a seismic environment and reduce the likelihood of a severe release.

Third, the prospects for cost effectiveness of a safety assurance program are excellent. PECO has searched for cost-effective major risk reduction features such as ATWS Alternate-3A features instead of Alternate-4A features. This resulted in some very highly cost-effective fixes. Although the residual risk is not known at this time, the comparatively low cost associated with studies and alteration to procedures or training suggests that such a program to maintain and harvest the insights of the PRAs for the conduct of operations are very likely to be cost effective. (For example, the staff previously cited the example of improving modification to the safety system's room cooler modification.)

Fourth, the staff believes that a large part of the value of the safety assurance program concept lies in the familiarity gained by operations personnel on how their responsibilities relate to (are important to) risk. To achieve this goal, it is important that the applicant integrate the program into its operations organization, and minimize the extent to which it is an external or contracted function.

Fifth, the NRC staff recognizes that PRAs contain few allowances for wearout. Future changes in the frequency of component failures, human errors, initiating events, precursors at other plants, or information from reactor safety research might turn up clues to higher risk. A followup program to maintain and improve the Limerick PRA/SARA studies could provide a mechanism to better assure that the lessons from such experience are thoroughly understood and, where necessary, acted upon. In a letter to the applicant dated June 7, 1984, the NRC staff addressed the key elements of the continued use of risk assessments during startup and operations and invited the applicant's response. The letter is included as Appendix B.

9.0 NRC STAFF CONTRIBUTORS AND CONSULTANTS

This report is a product of the NRC staff and its consultants. The NRC staff members listed below were the principal contributors to this report. A list of consultants follows the list of staff members.

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APPENDIX A

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APPENDIX B

SAFETY ASSURANCE PROGRAM

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

JUN 7 1984

Docket No. 50-352/353

Mr. V. S. Boyer, Sr. Vice-President
Nuclear Power
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Boyer:

SUBJECT: REVIEW OF THE LIMERICK RISK ASSESSMENTS

Members of the staff met with Philadelphia Electric Company representatives on March 30, 1984 to describe areas where our review of your risk assessments of the Limerick Generating Station has identified potential improvements. This letter is to affirm our comments during that meeting.

Over the past few years, our staff and consultants have met frequently to exchange information concerning the plant and the assessments of its risk in a high population density area. The attitude exhibited by Philadelphia Electric Company during these exchanges has been valuable to the conduct of our review. The public risks from the Limerick Generating Station have been reduced through design changes motivated by what the Philadelphia Electric Company learned during the performance of the Probabilistic Risk Assessments.

During the staff's review, we identified certain other potential improvements which were described at the meeting of March 30, 1984. The Philadelphia Electric Company indicated an interest in these potential improvements and subsequently provided comments (Letter to A. Schwencer, NRC, from J. S. Kemper, PECo, subject: Limerick Generating Station, Units 1 and 2 Risk Assessments, dated April 11, 1984). The Philadelphia Electric Company indicated that it was aware of the three specific improvements and had taken steps already to accelerate the scheduled ADS modifications, to mitigate HPCI and RCIC room heatup, and to train operators in the extended use of the containment sprays.

We all know that there are large uncertainties associated with the numerical results from current risk assessments. However, we believe that the potential improvements identified from the performance of the risk assessments transcend those uncertainties. Additionally, the process of conducting risk assessments provides a means of integrating the myriad of potential hazard considerations into a perspective that enables us to better discriminate what is of central importance to reactor safety from what is not of central importance. Such a tool provides a perspective that enhances communications among design and operational groups with different responsibilities or areas of expertise as well. I want to take this opportunity to reemphasize our interest in your maintaining and continuing to utilize the risk assessments during startup and operations. Some elements of this activity are described in Attachment 1.

We look forward to hearing your response to this preview of the results of our review in time to reflect any comments when documenting our review.

Sincerely,

H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Elements of Continued Use of Ris
   Assessments.

Attachment 1

Key Elements of Continued Use of Risk Assessments

1. Formal calculations of quantitative measures of importance to risk for initiating events, systems, components, human interactions in maintenance, surveillance, and operations. Such figures of merit bearing upon the importance of safety can be illuminating in several ways: a) they may reveal limitations in the PRAs; b) they are useful in the training of operators and maintenance personnel; and c) they are useful in the evaluation of procedures, technical specifications, and situations that may arise in plant operations.
2. Fault Hazards Analysis applied to hypothetical errors in the conduct of maintenance procedures, surveillance procedures, normal and emergency operating procedures, and technical specifications. This constitutes a formal "what if" examination of potential human error in the conduct of operations.
3. Where the importance-to-risk and the fault hazards analysis suggest that procedures may warrant improvement, the analysis can be extended to human error Failure Mode Effects Analysis. Changes, in procedures, technical specifications, operator training, system design, and/or control room simulator design as appropriate, suggested by such analyses, can be considered and where plausibly cost effective, be instituted.

4. Operations and maintenance personnel can be trained on the results of the studies into the importance-to-risk of their responsibilities, taught pattern recognition for the more vulnerable plant configurations, or circumstances and diagnosis of the more important accident scenarios.
5. From time to time the PRA quantification can be updated to reflect accumulated experience on the frequency of component failures, human errors, and initiating events. This effort can be made economical by employing the quantitative measures of importance-to-risk to assess the significance of altered fault event frequency, so that comprehensive and burdensome recalculations of risk are rarely necessary.
6. Philadelphia Electric, can devise and implement criteria spelling out thresholds for corrective action upon discoveries of less-than-expected system reliability, procedural adequacy, or greater-than-expected risk, where the Limerick PRAs serve as the frame of reference.
7. The risk assessment models can assess the risk importance to Limerick of precursor events that occur at other plants. Criteria can be established to determine the relevance of the events at other plants to Limerick in the sense of NUREG/CR-2497.

8. The results of the importance-to-risk evaluations can be made widely available, including the utility's quality assurance organization not only to enable reviews to be made, but for use in sharpening the focus or allocation of emphasis in the work of the QA audits.

9. The risk assessments and the assessments of importance-to-risk based upon it, can be maintained, and when appropriate, revised to make it a current, up-to-date evaluation tool.

Appendix C

REVIEW OF SEISMIC HAZARD AND FRAGILITY IN THE LIMERICK GENERATING STATION SEVERE ACCIDENT RISK ASSESSMENT

Appendix C

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1 INTRODUCTION

1.1 Background

The evaluation reported in this appendix beyond section 1.1 was completed in September 1983. The review of the earthquake portion of the LGS-SARA which follows indicated areas that could be improved by additional clarifications and studies. However, we believe that further studies would not change our basic conclusions that there is no obvious weak link in the seismic design of Limerick. These areas, outlined in the form of recommendations for further study in each of the following sections on seismic hazard, and structural, mechanical, component and equipment fragilities, address a wide range of specific seismological and engineering topics. In addition to performing PRAs considerable effort is ongoing in the seismic community in the assessment of risk related in part to these topics. If information becomes available that changes our conclusions regarding the significance of these recommendations, the staff would recommend additional studies to assess the effect of this information on the seismic risk posed by the Limerick Generating Station.

NUREG/CR-3756 dated April 1984 on the Seismic Hazard Characterization Program being carried out for the NRC staff by the Lawrence Livermore National Laboratory is discussed in Appendix D.

1.2 Introduction

An evaluation of risk due to seismic initiating events has been presented by the applicant, Philadelphia Electric Company (PECO), in the report entitled "Severe Accident Risk Assessment, Limerick Generating Station" (LGS-SARA). The NRC requested Brookhaven National Laboratory (BNL) to review the LGS-SARA. The NRC staff review of seismic hazard and fragility in the LGS-SARA, which follows, is based upon its own review, the BNL review, meetings with BNL, PECo and their respective consultants and the response to questions from the NRC submitted to PECo.

The LGS-SARA represents the third probabilistic risk assessment (PRA) submitted to the NRC which includes seismic initiating events. The previous submittals evaluated seismic risk at the Zion and Indian Point operating nuclear power plants. The LGS-SARA is submitted coincident with an operating license review that makes use of the extensive safety evaluation found in the Limerick Generating Stations Units 1 and 2 Final Safety Analysis Report (FSAR). The review concentrates on assessing the adequacy of the numerical values used in describing seismic hazard, fragilities and their associated uncertainties and gaining insight into the seismic capacity of the plants, particularly at levels beyond the safe shutdown earthquake. The applicant's prediction of core melt frequency resulting from seismic initiating events as calculated in the LGS-SARA is reported as having a median value of 3.3×10^{-7} per year with a 95th percentile value of 2.7×10^{-5} and a 5th percentile value of 2.2×10^{-9}. The applicant's point estimate of the mean annual frequency of a seismically induced core melt is reported to be 5.7×10^{-6}. The applicant indicates in the Response to NRC Questions that seismic events are not a major contributor to total core melt frequency. On the other hand, because of the specific nature of certain seismic sequences, they are major contributors to total early fatality risk.

By comparison, in the Zion and Indian Point PRAs it was calculated that seismic events are major contributors to both core melt frequency and risk. The NRC staff recognizes that the current state-of-the-art of seismic event probabilistic risk assessments is greatly constrained by the limitations of current knowledge and methodologies. The applicant's presentation of discussions of the many contributions to uncertainties in the LGS-SARA reflects their awareness of these limitations. Due to the unavailability of precise definitive procedures to quantify seismic risk, the current state-of-the-art requires the use of generalized probabilistic models which for the most part rely heavily on engineering judgements and expert opinion. As a consequence the resulting numerical estimates may contain unknown systematic biases and they may be most appropriately used in a relative rather than absolute manner. It is our judgement that the reliance upon the mean or median seismic risk estimate to reflect actual risk is premature. The wide bands of uncertainty presented in the LGS-SARA can be thought of as representing a large part, but not all, of the actual uncertainties. They may be used to gain insight as to the range of the actual risk associated with seismic initiating events at Limerick.

The LGS-SARA also affords evidence that the Limerick Generating Station Units 1 and 2 are capable of withstanding ground motion beyond the SSE and that there is no obvious weak link in their seismic design.

2 SEISMIC HAZARD

2.1 Scope

The probabilistic assessment of the seismic ground motion hazard at the Limerick Generating Station site was performed by Ertec for the applicant. The results of this study were incorporated with results of other studies to calculate the probabilities of structural, mechanical and equipment failure at the facilities. Our comments are based on our review of Chapter 3 and Appendix A of the Limerick Generating Station, Severe Accident Risk Assessment (LGS-SARA) report (Reference 1) and other relevant chapters related to the seismic hazard analysis in this report, on the review of our consultant Brookhaven National Laboratory (BNL) (reference 2) and upon our past experience in reviewing probabilistic estimates of earthquake hazard at other nuclear power plant sites such as Zion (Reference 3) and Indian Point (Reference 4).

2.2 Methodology

Th source of the earthquake data used in any Probabilistic Risk Assessment (PRA) is based on available earthquake catalogs. One of the fundamental problems with all methods of risk assessment is predicting hazard for extreme events at sites where little data exists and where the physical process of earthquake generation is not well known. For example, in the eastern U.S. where the Limerick plant is located, our knowledge of the input parameters required for probabilistic seismic hazard analysis is limited because of the short seismic history, unknown tectonics and lack of strong motion data.

The seismic hazard model used in the SARA study is described by Cornell (References 5 and 6) and McGuire (Reference 7). In both these approaches the calculations of seismic hazard are based on the definition of the following key parameters:

1. source region geometry,
2. earthquake recurrence model, and
3. attenuation model.

The source regions are defined on the basis of historical and instrumental seismicity, geologic and tectonic features. Observed seismicity and judgement are then used to estimate the recurrence statistics (a and b-values) and to describe the upper bound magnitude for each zone. Combining these with the attenuation model, the probability of exceedence as a function of acceleration is derived. Since there is much subjective judgement introduced in choosing some of the parameters used in the hazard model, there is a large measure of uncertainty in the final hazard estimates.

In general, in the LGS-SARA study, the methodology used is based on the state-of-the-art approach and is up to date. It is essentially the same methodology used in the Indian Point study, and can be summarized as follows.

1. In the LGS-SARA study four sets of seismogenic source zones were defined. These zones, as indicated by the applicant's consultant (ERTEC), cover a range of alternate interpretations around the LImerick site and they are:

   a. the Piedmont model,
   b. the Northeast Tectonic model,
   c. the Crustal Block model, and
   d. the Decollement model.

   For each model the seismicity of each zone was examined and the maximum historical earthquake in the zone was assigned. A subjective weight was then assigned for each set of zones.

2. Using the historical earthquake catalogs of Bollinger (Reference 8), Nuttli (Reference 9), and Chiburis (Reference 10); the rate of occurrence for each seismogenic zone was determined. Based on the Ertec assumption that small magnitude events rarely cause structural damage due to their short duration the activity rates were calculated for earthquakes with $m_b = 4.5$ and greater. Based on expert opinion (Reference 11), a b-value of 0.9 was used for all the seismogenic zones. Ertec used a single value indicating that uncertainty in the b-value has little effect on the hazard.

3. Based on subjective judgement Ertec assigned upper bound magnitudes for each of the seismogenic zones. For the dominating zones in the Piedmont, Northeast, Crustal Block and Decollement models they used 6.3 and 5.8, 5.0, 6.0 and 5.5, and 6.8 respectively as the upper bound magnitudes.

4. Using the Nuttli (Reference 12) formulation with adjustment for the anelastic attenuation constant in the northeastern U.S., sustained acceleration was estimated for the Limerick site. The effective peak acceleration (EPA) was assumed to be essentially equivalent to sustained acceleration.

   Due to the variation of the strong motion data a lognormal distribution was used to represent the uncertainty in the ground motion estimates. A value of 0.6 was used for ground motion dispersion (σ). Based on the assumption that Modified Mercalli (MM) intensities in a region are bounded, an upper bound cutoff acceleration for each region was assigned. These upper bound cutoffs are based on subjective judgement relating intensity to acceleration.

2.3. Evaluation

2.3.1 Seismic Zonation

The staff believes that the zones presented by Ertec are reasonable, except that the boundaries of some of the zones are not always drawn to coincide with well-defined geologic or seismologic features. There is great uncertainty, for example, in the Crustal Block and the Decollement models. The Crustal Block hypothesis is based on gravity and magnetic lineations and the assumption that these lineations define large blocks of earth's crust along which movement would occur generating earthquakes. In the LGS-SARA, Ertec identified eight zones to constitute the Crustal Block model. The staff cannot see the correlation between the seismicity distribution and all of the eight zones identified in the LGS-SARA report. The staff concludes that since the site lies in the Triassic basin as indicated in the FSAR (Reference 13), and since zone 8 of the Crustal Block model represents the same Triassic basin, the Limerick site should be included within zone 8. The staff estimates that variations in the boundaries in zone 8 of the Crustal Block model would contribute to changes in the final results of the hazard. Similarly, definition of the Decollement zone boundary is not well defined. The Decollement zone was hypothesized to account for the generation of large earthquakes such as the Charleston, South Carolina earthquake of 1886. Although no clear evidence thus far supports this hypothesis, a probabilistic study may consider the Decollement as a source of large earthquakes.

The staff does not agree with the subjective weights assigned to the different zones. For example, although the Crustal Block model is not well defined and not well recognized in the scientific community it is weighted equally with the more accepted Piedmont and Northeast Tectonic models.

The applicant solicited expert opinion with regard to the weight that should be assigned to a tectonic hypothesis (not necessarily the Decollement model) that would allow the occurrence of a large magnitude earthquake ($M \cong 7$) near the site. The four experts chose weights varying from 0.0 to 0.3. BNL believes that the 0.1 weight assigned by the applicant should be considered as a lower bound.

Alternatively, BNL proposed that a large earthquake such as the Charleston event should additionally be considered for each of the four seismogenic zones proposed by Ertec. The staff does not completely support this hypothesis unless the seismotectonic zones are redefined to reflect seismicity patterns or tectonic features which may accommodate a Charleston type earthquake. If such a large event (for which the applicant assumes no upper bound to effective peak acceleration) was considered, the hazard curves would be unbounded and they would contribute to an increase in the core melt frequency.

2.3.2 Seismicity Parameters:

Ertec used 0.9 for the b-value based on expert opinion. The staff concludes that an appropriate approach would be to use the available data from the different catalogs in the eastern U.S. and try to estimate a b-value which is more characteristic for the zones around Limerick than the one used in this study. No uncertainty in the b value was considered in the Ertec study. It was simply stated that b-value would not have great effect on the seismic hazard and as a result the statistical uncertainty in the b-value was ignored.

The other parameter needed for estimating the seismic hazard is the upper bound magnitude for each of the seismotectonic zones. Regarding the Charleston-type earthquake of magnitude 6.8, Ertec did not provide any supporting evidence that this earthquake is the maximum event that could occur on a Decollement type structure. A sensitivity study would have shed some light on the effect of variation in the upper bound magnitude on the hazard curves.

2.3.3 Ground Motion Attenuation

Using Nuttli's approach, sustained acceleration was estimated after adjusting for the attenuation factor by using a more representative value for the northeastern U.S. than the original value used for the central U.S. Using a factor of 1.23 Ertec then converted sustained acceleration to effective peak acceleration (EPA). The definition of EPA is a highly controversial issue. Additional studies supporting the applicant's definition of this parameter and the sensitivity of the results to its use would be helpful. For estimating the seismic hazard, a lognormal distribution about the mean value was used, with a value of $\sigma = 0.6$. This value is based mainly on western U.S. data. Whether or not the value is applicable to the eastern U.S. is a question which needs further investigation. The statistical scatter about the mean attenuation relationship characterized by σ is an important parameter in the seismic hazard analysis. The influence of σ on EPA estimates is very significant particularly for low probabilities. It is difficult to assess an appropriate value of σ for use in Eastern U.S. seismic risk analysis, therefore a sensitivity study regarding this factor would provide more insight on its contribution.

Also, the choice of upper bound cut off to effective peak acceleration is highly judgmental. There is insufficient evidence to support this choice. A sensitivity study regarding this parameter would provide some insight on its effect on the hazard curves.

2.4 Comparison of LGS-SARA with Indian Point PRA

The staff compared the seismic hazard generated for the LGS-SARA with that for the Indian Point PRA. The comparison showed little difference between the ranges of the hazard curves for the two sites. From examining the seismicity of the region, we expected a greater difference between the two seismic hazard studies due to the higher seismic activity around Indian Point. However, since LGS and Indian Point were generally assumed to lie in the same seismogenic source zone and since the seismic activity in these zones is considered to be uniformly distributed, the differences in the computed hazard were minimized. The small differences that do exist between the two sets of hazard curves can be attributed to small assumed differences in the area of the seismotectonic zones considered in the two sutdies, the upper bound magnitude, and the activity rate considered for each zone.

2.5 Conclusions

The methodology used in the LGS SARA report to estimate the seismic hazard is adequate and the approach is well established. It has been used before for the Zion and Indian Point PRAs.

Although Ertec used their best judgement in defining the different parameters used in the model, the staff has a few concerns regarding these parameter definitions.

1. The uncertainty associated with the Crustal Block model was underestimated. The Limerick site should have been included within zone 8 of the Triassic basin in the Crustal Block model. With regard to the Decollement zone, a

higher weight and/or alternative models which allow a large magnitude earthquake to occur near the site should have been examined.

2. The uncertainty associated with the two significant parameters (b-value and ground motion dispersion σ) should have been considered.

3. A single upper bound magnitude for the Decollement was used without justifying its uniqueness.

4. Uncertainty was not considered in the choice of the upper bound cutoff to effective peak acceleration.

The staff recommends that although peak ground acceleration is an appropriate measure of damage over certain frequency ranges, the probabilistic analysis should more directly estimate ground motion at all relevant frequency ranges of the spectrum.

Finally, the BNL review states that although the effect of the individual issues raised above on the mean frequency of core melt is judged to be small (less than a factor of 2), the total effect could be moderate (2<factor<10). Consideration should be given to verify that indeed this is the case.

In conclusion, the staff considers the general methodology used in this study to be appropriate. It should be stated, however, that because of the extensive use of subjective input and the uncertainty this engenders, the results of these studies are more appropriate for relative comparisons than for absolute determination of hazard and the resulting risk.

2.6 References

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3 GEOTECHNICAL ENGINEERING

3.1 Scope

The following summarizes the NRC staff's preliminary review of geotechnical engineering aspects of the Limerick Generating Station Units 1 and 2 Severe Accident Risk Assessment (LGS-SARA). Specific LGS-SARA elements reviewed include a) the applicant's consideration of geotechnical engineering related potential failure mechanisms resulting from a seismic event, b) the applicant's treatment of the geotechnical engineering aspects of the seismic hazard analysis, and c) the applicant's treatment of geotechnical engineering parameters affecting fragility analysis. The review was conducted in accordance with the applicable general guidance contained in NUREG-2300 Chapter 10, 11, and 12 (Ref. 1). This review also considered appropriate elements of the draft report of the preliminary review of the LGS-SARA performed by the NRC's consultant, the Brookhaven National Laboratory (Refer. 2).

3.2 Geotechnical Engineering Related Site Data

1. General Site Description - Topography of the Limerick site area consists of gently rolling ridges dissected by the courses of the Schuylkill River and its tributaries. The main plant structures are on a broad ridge approximately 100 feet above the river. The plant site is divided into three main subareas: (1) the reactor/turbine area at grade elevation 217 feet msl, (2) the cooling tower area at grade elevation 257 to 265 feet msl, and (3) the spray pond area with bottom of pond at elevation 241 feet msl and a normal still pond level of 251 feet msl. Ground water in the site area decreases from approximately 250 ft msl northeast of the spray pond area to an elevation of less than 120 feet msl southwest of the reactor/turbine area.

   Detailed descriptions of the geotechnical engineering aspects of the general site ar- are presented in Limerick FSAR Sections 2.5.4 and 2.5.5 (Ref. 3). The bedrock in the site area consists of interbedded red sandstones, siltstones, and shales indurated to a depth of several thousand feet, which are moderately to closely jointed. Bedrock in the immediate plant area dips 8 to 20 degrees to the north. The soils at the site consist of red sandy and clayey silts with rock fragments derived from weathering of the underlying bedrock. Soil thickness ranges from 0 to 40 feet, averaging 10 to 15 feet. For the most part, soils below a depth of 10 feet consist of highly weathered and fractured rock with intermixed silts and clays.

   The main seismic Category I plant structures including the reactor enclosure, control structure, diesel generator enclosure, spray pond pump house, spray pond spray network, turbine enclosure and radwaste enclosure are founded on unweathered bedrock. Seismic Category I facilities not founded completely on bedrock are founded totally or in part on natural soil or manmade fill and include the diesel fuel oil storage tanks, buried cooling water piping, a pipe valve pit, electrical ducts and the northwestern portion of the spray pond.

2. Properties of Site Subsurface Materials

The Limerick site investigation program, which was accomplished between 1969 and 1977, was accomplished in a deterministic manner. The applicant has reported in the PSAR (Ref. 4) and FSAR (Ref. 3) that site exploratory investigations included 380 borings, 17 test pits, and 10 seismic refraction traverse lines, totalling 5180 linear feet. In addition, a surface shear wave velocity survey, a seismic uphole survey, inhole permeability testing, plate bearing testing, micromotion measurement testing, and in situ bedrock stress testing were accomplished in the site vicinity.

As sampling theory was not used in the development of the site investigation program and as the number of samples of rock and soil materials tested would be considered small in a statistical sense, the NRC staff concludes that the application of probability theory to determine statistically significant estimates of means, variance, and probability density functions for pertinent soil and rock properties at the Limerick site is neither justified nor appropriate for use in the LGS-SARA analyses.

Therefore, it is the conclusion of the NRC staff that the use of deterministically estimated representative rock and soil material properties values in the LGS-SARA is acceptable.

3.3 Failure Mechanisms

In the LGS-SARA, the applicant addressed earthquake-induced acceleration as the potential failure mechanism capable of producing structural and component failures at the Limerick site. Other potential failure mechanisms, including subsidence and acceleration-induced liquefaction and settlements were not explicitly considered.

1. Subsidence - The NRC staff review of site data contained in the Limerick PSAR (Ref. 4) and FSAR (Ref. 3) presented no evidence of zones of solutioning, caverns, or highly weathered areas in the foundation bedrock or soils which would allow significant subsidence under any proposed seismic loading. The NRC staff therefore concurs with the applicant's exclusion of subsidence as a probable failure mechanism requiring consideration in the LGS-SARA.

2. Liquefaction - The probability of failure of structures, systems, and components under seismic loading conditions due to liquefaction of foundation and backfill soil was not explicitly addressed in the applicant's report. Based on the site data presented by the applicant in the FSAR, the NRC staff independently analyzed the liquefaction potential of the natural residual soils and backfill materials at the Limerick site and finds the following:

   a. The natural residual soils at the site evidence an average SPT resistance of 46 blows per foot, exhibit cohesive characteristics (as evidenced by an average plasticity index of 8), and in general can be considered to have a low potential for saturation due to the relatively severe water table gradient at the site (Refs. 4 and 3). The NRC staff considers that such soils have a negligible potential for liquefaction.

b. The applicant reported in the FSAR that all fills associated with seismic Category I structures and piping were classified as either mass concrete fill, cementitious backfill, select granular backfill, or Type 1 random fill. The mass concrete fill and the cementitious backfill were batched to attain a 28-day compressive strength of 2000 psi and 80 psi, respectively. Such materials are not capable of liquefaction. The select granular backfill material consists of 3/4-inch maximum size aggregate with less than 10% by weight passing a No. 200 sieve and was compacted to 95% AASHTO T-180 maximum dry density. The Type I random fill consists of 8-inch maximum size broken rock graded course to fine and was compacted to 90% AACHTO T-180 maximum dry density (Refs. 4 and 3). Due to the relatively dense nature of the select backfill and Type I random fill material, the NRC staff considers these back fill materials would not be susceptible to liquefaction under the seismic loading conditions postulated for the Limerick site.

The NRC staff therefore concludes that the potential for liquefaction of the Limerick site soil and backfill material due to postulated seismic loadings may be neglected without significantly influencing the overall LGS-SARA results.

3. Settlements - The applicant has not explicitly analyzed rock and soil settlement and differential settlement as potential failure mechanisms in the LGS-SARA. In the Limerick FSAR the applicant has deterministically estimated a maximum settlement of the reactor building, the heaviest structure at the site, to be on the order of one quarter of an inch or less due to pseudo-elastic compression of the rock occurring upon application of loading with construction (Ref. 3). The NRC staff has independently verified these findings using the procedures in Reference 5 and has further considered the potential for total settlement under possible seismic loadings. Considering pseudo-linear-elastic response and input seismic accelerations up to 4 times the SSE, the NRC staff has deterministically estimated an upper bound total rock deformation of less than 0.5 inches. The NRC staff therefore considers that there is a negligible low likelihood of rock settlement becoming a failure mechanism that would require assessment. Consideration should however be given to verifying that potential failure of safety-related piping and of small lines attached to safety-related piping near the junction of the containment building and the reactor enclosure due to impact or relative displacement of the buildings will not contribute to the frequency of core melt (Ref. 2).

   The NRC staff has also analyzed the potential for settlement of residual soils and backfill materials due to a seismic event. Using the procedures of References 6 and 7, the staff concludes that the maximum upper-bound settlement of soils and backfill materials supporting seismic Category structures systems on components would be expected to be less than 1.0 inch for seismic loadings up to 4 times the design SSE. The NRC staff considers that there is very little likelihood that soil settlement or differential settlements due to seismic events of this magnitude could become a significant failure mechanism for structural systems and components founded on soils.

The NRC staff therefore concurs with the applicant's exclusion of settlements and differential settlements as significant potential failure mechanism requiring detailed analyses in the LGS-SARA.

3.4 Seismic Hazard Analysis

The procedures used by the applicant in the development of the Limerick seismic hazard model do not explicitly consider the specific Limerick site soil and rock engineering properties. In his analysis the applicant used an attenuation function to estimate peak ground accelerations that was developed from an analysis of existing recorded strong motion data. The function was derived from a regression of peak ground acceleration against magnitude and distance and assumes a regionally constant anelastic attenuation factor (Q). Local site characteristics relating to the geometry and engineering characteristics of the near surface soil and rock materials associated with the strong motion records were not separately accounted for in the regression. Uncertainties in the peak ground accelerations predicted through the use of this function attributable to local site conditions are therefore combined with those associated with source and propagation path effects. In applying the developed attenuation function to the Limerick site the applicant assumed a lognormal distribution of acceleration about the mean value with a standard deviation of 0.6 selected as typical of the scatter associated with strong motion data sets from a specific geological region. This standard deviation value corresponds to a factor of 1.8 times the median value.

There are no geotechnical engineering related local site features known to the NRC staff which would preclude considering this site to be within the geologic regional average for which the applicant-developed attenuation equation is intended to apply. The NRC staff recognizes that the physical processes affecting local site response are not well understood. The NRC staff is also aware that large uncertainties due to source and path parameters as well as local site-specific geotechnical engineering related parameters are already reflected in the variance in the data used to estimate peak ground acceleration. The staff therefore considers that a vigorous analysis of the influence of local soil and rock property parameters on the attenuation of acceleration at the Limerick site is not warranted nor appropriate in keeping with the general level of the state of the art in predicting ground motion in the eastern U.S.

3.5 Fragility Analysis

1. Structures, Systems, and Components Founded on Rock

   In the LGS-SARA the seismic fragility of structures, systems, and components founded on rock were described in terms of the median ground acceleration capacity associated with seismic-induced failure and of the logarithmic standard deviation of this median value. As an aid to computation, the applicant used an intermediate variable called the "median factor of safety." It was defined as the ratio of the estimated median ground acceleration capacity causing failure to the Safe Shutdown Earthquake (SSE) acceleration used in the design analysis. Thus, rather than directly estimating the seismic fragility of structures and components founded on rock, the applicant estimated median factors of safety and logarithmic standard deviations against failure based upon the deterministically accomplished design response analysis. In the deterministic design of the Limerick Plant, two-dimensional lumped mass models were developed for the major seismic Category structures founded on rock. Separate models were developed for the north-south, east-west, and vertical responses analyses for each structure. Because of the relatively high stiffness of the rock the applicant treated the foundation rock as a fixed boundary for the analysis of all structures excepting the primary containment and the reactor enclosure and control structure. The floor response spectra developed for these two structures for equipment analysis purposes were based upon models considering the elastic deformation of the supporting medium. The shear modulus, shear wave velocity, and the density of the supporting rock used in the analysis were 1.2×10^6 psi, 6000 ft/sec., and 150 lbs/ft^3 respectively. (On page 4-22 of Appendix B of the LGS-SARA the reported bedrock modulus of elasticity of 7.3×10^6 psi is in error. Revision 19 to the LGS FSAR corrected that value to 3.0×10^6 psi to reflect the value actually used in design (Ref. 3)). Embedment conditions were neglected in the design analysis. To account for variations in the structural responses owing to uncertainties in the material properties and to approximations associated with the modeling techniques used in the design analysis, the computed floor response spectra were smoothed and the spectra were broadened on either side of the peak value by 15% of the frequency at which the peak occurred. Additional soil structure interaction analysis were performed to access the sensitivity of the design models to variation in rock modulus. Model analysis demonstrated that for a variation in rock modulus of ± 50 percent, variations in structural frequencies did not exceed 10 percent for predominant modes.

In the LGS-SARA seismic fragility analysis the applicant did not explicitly consider geotechnical engineering parameters beyond those in the deterministic design analysis. The applicant treated the uncertainty introduced into the calculated design response of structures due to variability in geotechnical engineering related parameters only by including it as an element of all randomness and uncertainty associated with soil-structure interaction effects. No uncertainty was assigned to the ground response spectrum factor used in the analysis due to variation in foundation material properties. Lacking quantitative evidence from site specific sensitivity analyses data to estimate the variability in the median factors of safety for structural capacity due to geotechnical engineering related parameters required to define the fragility curves for the plant structures, the applicant used subjective engineering judgment. In the applicant's judgment the major plant seismic Category 1 structures are considered to be founded on competent rock and the design of the structures was conducted using assumptions and methods of analysis that result in small variation in frequency and response when significantly large variations in the flexibility of rock and of energy dissipation in rock by radiation damping are considered. The applicant therefore concluded that the design results would have a median factor of safety of 1 based upon soil structural interaction considerations. Using similar reasoning and considering the nature of the model used in the deterministic design, the applicant assigned a relatively small logarithmic standard deviation of 0.05 to the uncertainties in the median factor of safety for the overall structural acceleration capacity due to all soil structure interaction effects.

The NRC staff considers that although the deterministically derived design structural response to the SSE cannot be accepted as an "absolute best estimate" of a median value for structural acceleration capacity when considering geotechnical engineering parameters at the Limerick site, it is an acceptable estimate considering the level of effort and the analytical model used. In the applicant's methodology the individual uncertainties associated with each factor bearing on the variance of the mean structural capacity are summed to obtain an overall logarithmic standard deviation using the "square root of the sum of the squares" process. In this process, because of the number of factors considered and the relative size of the uncertainties for each factor, the addition of reasonable amounts of uncertainties to a few factors would result in only a very small increase in the overall summation. The NRC staff therefore concludes that although the applicant has not incorporated the total effect of variation due to geotechnical engineering related factors into the total overall structural acceleration capacity by the procedures used, the inclusion of a reasonable additional uncertainty value for geotechnical parameters would only produce a small effect which would not be significant in the final product of the analysis (Ref. 2).

The NRC staff also considers that neglecting embedment considerations in the basic design tends to bias the "best estimate" of the median structural capacity to the conservative side especially at higher acceleration levels. This conservatism is acceptable to the NRC staff. However, because embedment conditions were neglected in the original deterministic design the effect of soil pressure on buried walls expressed in terms of variance of the mean factor of safety for capacity, were not explicitly addressed in the LGS-SARA. Consideration should be given to evaluating the impact of embedment on the fragility of affected seismic Category I walls and any supported systems or equipment under greater than SSE loadings (Ref. 2).

2. <u>Structures, Systems, and Components not Founded on Rock</u>

The applicant did not address fragility of structures, systems, and components which are partially or totally founded on soils in the LGS-SARA. Seismic Category I facilities not found completely on competent bedrock include the diesel oil tanks, underground piping, a piping value pit and electrical ducts. These structures, systems, and components have been evaluated by the applicant deterministically for an SSE of 0.15g in conjunction with the Limerick Safety Evaluation. Soil response studies were performed by the applicant using the computer program "Shake" to estimate ground motion induced by a safe shutdown earthquake in the backfill material surrounding and supporting buried seismic Category I piping system. Earthquake motion was specified at the level of the top of rock and resulting peak accelerations were computed at the level of the pipe. The sensitivity of output to variation in soil shear modulus was also considered. The applicants reported results indicate an approximate 2 fold amplification of input acceleration results when input accelerations are equal to the SSE.

Since the response of soil to seismic input motion is nonlinearly strain dependent consideration should be given to verifying that soil supported safety-related piping, and other soil supported structures and components are not stressed to the point that they would significantly contribute to the frequency of core melt when considered to be exposed to seismic events greater than the SSE.

3. Spray Pond

The fragility of the seismic Category I spray pond which provides the ultimate heat sink for cooling water was not addressed by the applicant in the LGS-SARA. Based upon a review of information presented by the applicant in References 3, 4, 8, and 9, the NRC staff has evaluated the stability of the spray pond slopes. The slopes of the ultimate heat sink spray pond were excavated partly in soil and partly in rock. The applicant has deterministically designed the spray pond slopes using protection riprap stone materials with stone size, layer thickness, and slope geometry governed by the anticipated wave conditions expected during the Probable Maximum Flood (PMF). In addition the applicant has deterministically analyzed the stability of the spray pond slopes to demonstrate the stability of the soil and rock slopes under the design basis conditions of an SSE of 0.15g. The NRC staff concluded that the scope of the applicant's field and laboratory efforts was adequate to define the bedrock and foundation conditions at the spray pond site and to establish appropriate deterministic design basis strength parameters of the slope materials. The NRC staff also found the rock and soil slopes acceptably stable under a design basis SSE of 0.15g.

The applicant has not presented an analysis of the effects of seismic loading greater than 1 times the design SSE on the stability of the spray pond slopes and of the water holding capability of the spray pond after such seismic events. However, it is the NRC staff judgement that considering the configuration of the spray pond, the topography of the site, and the geometry and strength of the rock and soil slopes, the impact would be small for acceleration levels up to 2 times the SSE. Uncertainties associated with the strain dependent non-linear response of the soil slopes of the spray pond founded above near surface bedrock, preclude fragility judgements when greater input accelerations are considered. Consideration should therefore be given to accurately defining the impact of the hypothesized exposure of the spray pond soil and rock slopes to seismic events 2 to 4 times the design SSE on the overall core melt frequency.

3.6 Conclusion

The NRC staff review of the LGS-SARA indicates that the report of the applicant did not explicitly address geotechnical engineering parameters impacting upon core melt frequency. The staff's evaluation of the LGS-SARA, however, indicates that the methodology used by the applicant in the seismic hazard and seismic fragility analysis for the most part adequately envelops geotechnical engineering parameters considering the state of the art of the methodology and the large uncertainties associated with the overall analysis.

The NRC staff review of the LGS-SARA also found that failure mechanisms relating to geotechnical engineering parameters other than acceleration, i.e., subsidence, liquefaction, and settlements were not explicitly addressed in the LGS-SARA. Using the site data presented by the applicant in the PSAR and FSAR, the NRC staff analyzed the potential for occurrence of these failure mechanisms in the soil and rock areas of the site. The results of this analysis indicate that there is a negligibly low potential for structures, systems, and components failures due to possible effects of these mechanisms. The NRC staff therefore concurs with their exclusion from consideration in the LGS-SARA.

Based upon the NRC staff and consultants review of the information presented in the LGS-SARA the following items related to geotechnical engineering aspects of the review are presented for consideration

(a) Although settlement and differential settlements of structures are not considered to be viable failure mechanisms for the Limerick site requiring comprehensive treatment in the LGS-SARA, consideration should be given to verifying that potential failure of safety-related piping and of the small attached lines located near the junction of the containment building and the reactor enclosure caused by impact or relative displacement of the buildings will not contribute to the frequency of core melt.

(b) Because embedment condition were neglected in the original design, the effect of soil pressure on buried walls expressed in terms of variance of the median factor of safety for structural capacity was not explicitly addressed in the LGS-SARA. Consideration should be given to evaluating the impact of embedment on the fragility of affected walls and supporting systems.

(c) Because of soil amplification considerations related to structures founded upon soil, consideration should be given to evaluating the fragility of soil supported safety-related piping and other soil supported structures at seismic levels greater than the SSE.

(d) Due to geotechnical engineering related uncertainties associated with the capability of the spray pond slopes to withstand seismic loadings greater than about 2 times the SSE, consideraton should be given to accurately defining the fragility of the spray pond at SSE levels greater than 2 times the SSE.

3.7 <u>References</u>

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9. Geotechnical Engineers, Inc., "Report of Soil Testing - Limerick Nuclear Station Spray Pond, Winchester, Mass.," September 1974.

4 STRUCTURAL FRAGILITIES

4.1 Scope

The review comments and evaluation presented here are based on the preliminary review of those portions of Section 3.0 and Appendix B of the Limerick Generating Station Units 1 and 2 Severe Accident Risk Assessment (LGS-SARA) which are related to structural response and structural fragility formulation for a seismic event. The major aims of this preliminary review are:

(a) to identify, where possible, sources of conservatism and nonconservatism;

(b) form general impressions regarding the adequacy of the approach used of the findings of the LGS-SARA;

(c) to identify key contributing structural components, if any;

(d) compare the LAS-SARA with recent probabilistic risk assessments (PRAs) for other plants, if applicable; and

(e) to gain insights regarding probable seismic capacity of the plant beyond SSE.

In this review, the findings of the draft report prepared by the Brookhaven National Laboratory (BNL) (Reference 1) are relied upon heavily. The earlier review report prepared by Sandia National Laboratory (Reference 2) for the Indian Point Probabilistic Safety Study (Reference 3) is also relied upon extensively in this review. Additional information obtained from the licensee and its representatives in a meeting of August 5, 1983 is also reflected in this review findings.

4.2 Methodology of LGS-SARA

A brief description of the methodology used for developing structural fragilities in the LGS-SARA is given in this section. Structural fragility data are presented in the form of fragility curves which plot the fraction of expected failures versus effective peak ground acceleration. In Fig. 1, an example of a fragility curve is shown. In the LGS-SARA, generally, Seismic Class I Structures are considered to fail functionally when inelastic deformations of the structure under seismic load are estimated to be sufficient to potentially interfere with the operability of safety related equipment attached to the structure. Thus, the conditional probabilities of failure for a given free field ground acceleration for Class I structures are for operability limits and should not necessarily correspond to structural collapse.

In order to obtain fragility curves, the approach adopted in assigning capacities (failure fraction as function of effective peak ground acceleration) for the structures was to first determine the median factor of safety against failure and its statistical variability under the safe shutdown earthquake

(SSE). Then the median effective ground acceleration causing failure was estimated by multiplying the SSE acceleration level by this factor.

The overall safety factor was determined by evaluating the safety factors for a number of parameters, which fell into two categories: structural capacity and structural response. Parameters influencing the factor of safety on structural capacity include the strength of the structure compared to the design stress level and the inelastic energy absorption capacity (ductility) of a structure to carry load beyond yield. In the LGS-SARA, an additional parameter, earthquake duration factor, is also included in computing the median factor of safety on structural capacity. The parameters in structural response for a given ground acceleration are made up of many factors. The most significant of these include: (1) ground motion and the associated ground response spectra for a given peak free field ground acceleration, (2) energy dissipation (damping), (3) structural modeling, (4) method of analysis, (5) combination of dynamic response modes, and (6) combination of earthquake components. The derivation of each factor of safety considered variability. In each case, a median safety factor was assigned along with a variability. When combining the median safety factors of contributing parameters, their variabilities were also combined to define the overall safety factor. From this overall safety factor, the median effective or sustained peak ground acceleration associated with failure was determined as explained earlier.

The entire fragility curve for any structure can be expressed in terms of the best estimate of the median ground acceleration capacity and two random variables, one representing the inherent randomness of the event (B_R) and the other corresponding to uncertainty associated with predicting response to an event (B_U). B_R by definition is irreducible. For example, it is not possible, at least in the foreseeable future, to predict the exact time-history of an earthquake event at a given site, assuming that the occurrence of the event can be predicted. B_U, in a sense, represents a measure of our lack of knowledge for example, the mathematical modeling of a structure to predict the responses to a seismic event. As our knowledge advances, this uncertainty can be reduced.

In Fig. 1 an example of log-normally distributed fragility curve is shown. The solid curve is, effectively, the median fragility curve incorporating inherent randomness uncertainty, B_R. The left and right dashed curves represent certain percentile curves to reflect the uncertainty (B_U) in the median curve.

Note that seismically induced failure data are generally unavailable for structures. Therefore, each factor of safety and its variability and hence the final fragility curves are developed primarily from analysis and engineering judgment supported by limited test data. Table 1 lists the key structural components with associated capacity data in terms of median acceleration capacity and associated log-normal standard deviations.

The earthquake duration factor used in LGS-SARA has not been used explicitly in other PRAs. This factor, according to LGS-SARA, reflects the additional capacity due to the shorter duration with correspondingly lower energy content and fewer strong motion cycles present in the Limerick median expected earthquake as compared to the earthquake which would generate the number of cycles used in the determination of the median factor of safety related to the ductility of the structure.

It should be noted that the methodology used in the LGS-SARA, as in other PRAs, does not include an explicit consideration of design and construction errors and, hence, may be biased (Reference 1).

4.3 Evaluation Findings

In this section, review comments are presented on both general methodology and the key structural components listed in Table 1.

The following seismic Category I structures were evaluated in the LGS-SARA.

Primary Containment Structures
Reactor Enclosure and Control Structures
Spray Pond Pump Structure
Diesel Generator Enclosure
Spray Pond

It was judged in LGS-SARA that the failure of non-Category I buildings would not affect the seismic capacities of the Category I structures and, hence, fragility evaluation were not conducted for the non-Category I structures as part of this evaluation.

It is our understanding that the selection of the critical structural components was based on the identification by NUS of system and components important to safety and a plant walk-down performed by SMA. As indicated in Reference 1, since the plant is still under construction, a systematic review of the potential for secondary components failing, falling, and impacting primary components was not undertaken. Therefore, we concur with the recommendation in Reference 1 (p. 4-7) to conduct a systematic review of this aspect after the construction of the plant is completed.

4.3.1 Comments on General Methodology

The methodology used in the LGS-SARA is very similar to the methodology used in other PRAs (e.g. Reference 3), and as such is a state-of-the-art approach. However, the methodology is based on simple probabilistic models and hence, in our opinion, contains large uncertainty due to methodology itself. Specific comments on the methodology are as follows.

(a) The multiplicative model (i.e. the median of the overall factor of safety is a product of the median factors of safety for each variable) proposed in the section 2 of Appendix B of the LGS-SARA requires the mathematical condition that each median factor of safety be an independent variable. The licensee in a meeting and in a subsequent letter (Reference 4) indicated that the use of the above model is not intended to imply that each of these variables are totally independent. The estimated influence of dependency was considered in developing the factors of safety and log normal standard deviations for each variable. The applicant further stated that the overall median factor of safety and associated variabilities are checked for reasonableness for each structure and mode of failure.

Considering the state-of-the-art, the methodology is reasonable when the above fact is taken into consideration and the evaluation is performed by an experienced engineer. However, it must be noted that the above methodology has not been verified by either analytical investigation or adequate test data and, therefore, contains a great deal of uncertainty.

(b) The explicit use of the factor of safety associated with the expected duration of the earthquake is unique to LGS-SARA. In other published PRAs, the duration effect is accounted for by considering it to be incorporated in the concept of effective peak acceleration in the hazard estimation. In the LGS-SARA, the duration factor is considered independently and in addition to the use of effective peak acceleration. Reference 1 contains a detailed discussion of this factor including its effects on the risk results. The licensee provided additional information (Reference 4) to indicate that when the three factors of safety of (effective peak acceleration, ductility, and duration are considered simultaneously, the combined median factor of safety and uncertainty values are reasonable as compared to other PRAs.

(c) As discussed earlier, the design and construction errors are not accounted for explicitly in the fragility development. It is recognized that this is the limitation of the current state-of-the-art.

(d) As discussed in Reference 1 and reviews of other PRAs (e.g., Reference 3), additional studies and research efforts are required to justify the use of ductility factor for single degree of freedom models to represent multidegree of freedom structures. We concur with Reference 1 that higher uncertainty value should be assigned to this factor.

(e) We concur with the Reference 1, that uncertainty in some of the parameters has been understated (particularly, modeling uncertainties). The median capacity value may be on the high side in some cases.

(f) As in other published PRAs, LGS-SARA does not contain sensitivity analyses to indicate the robustness of the assumptions. However, in a August 5 meeting the licensee provided some discussion on the results of a recent sensitivity study which examined, for example, the effects of the assumption of different distribution (other than log-normal). We have not reviewed the results of this study. Reference 1 has included some sensitivity studies for some assumptions. It appears that effects of the assumption used in the developing fragilities LGS-SARA on seismic risk are minor.

(g) It was not clear to us whether or not dynamic lateral earth pressures were considered in the structural fragility evaluation. In a response to (Reference 4) the staff inquiry, the licensee stated the following:

"The Limerick structures are generally embedded in rock with lean concrete backfill. The rock and concrete backfill are separated from the structures by several inches of rigid insulation so that essentially no lateral loads can be transmitted to the structure from the rock. The seismic models, which were developed for the design analysis and which were used for the capacity evaluations, reflect this separation. No lateral loads are transferred from structure to rock or vice versa except at the base slab, and all shears and moments developed in the structure are transferred down to the base slab rather than being taken out at higher elevations.

Any local soil loads on the walls were judged to be small in comparison to the out-of-plane capacities of the walls, and therefore no reduction in the seismic capacities of the Limerick structures was judged appropriate. There is no evidence that dynamic soil pressures have ever failed basement walls unless gross soil failures have occurred. Such a failure is considered incredible at the Limerick site."

It is recommended that the above judgment should be verified by a specific analysis of the embedded portion of a reactor enclosure wall which takes into account the effects of soil pressures, where applicable.

(b) We concur with the Reference 1 that the implications of impact between the containment building and the reactor enclosure should be addressed for the following concerns:

a. Failure of safety-related electrical and control equipment located in the reactor enclosure.

b. Failure of safety-related piping which crosses between the two buildings due to relative displacements.

In additon, it should be verified that no safety-related components will be damaged by spilled concrete caused by impact of the two structures.

Finally, it should be verified that failure of small lines attached to the safety-related piping near the junction of the two structures and anchored to the reactor enclosure will not contribute to the frequency of core melt.

Based on the above discussion of the general methodology, it is apparent that current state-of-the-art for seismic risk evaluation precludes the determination of "absolute" seismic risk and it only provides a relative measure of risk between different sites and plants provided the methodologies and assumptions are consistent in each risk evaluation.

4.3.2 Comments on Critical Structures/Components

Table 1 lists the critical structural components or components which are affected by the estimation of structural response parameters. We have not performed a detailed review of the calculations for each of these critical components. However, we have reviewed the findings reported in the Reference 1 and the discussions in LGS-SARA.

(a) Condensate Storage Tank (S_2)

We concur with the findings in the Reference 1 that the fragility parameters for the condensate storage tank appear to be reasonable.

(b) Reactor Internals (S3), CRD Guide Tube (S5), Reactor Pressure Vessel (S6)

In the fragility estimation for these three components, a value of 10% damping was assigned to concrete portions of the support structure (also see Reference 1).

In Reference 4, the licensee quoted results of a recent study (Reference 5) which indicates median dampings at various stress levels. Based on these damping values, the licensee performed an analysis to indicate that composite damping value to be between 9 and 10 percent for the RPV support system at the median RPV fragility level. Provided the values in Reference 5 are acceptable, the issue of the damping value in concrete structure can be considered resolved.

We concur with Reference 1 that, as for other components, the modeling uncertainties are underestimated. According to Reference 1, the effect of doubling the uncertainty for modeling would have a small affect in the frequency of core melt.

(c) Reactor Enclosure and Control Structures (S4)

In Reference 1, it is estimated that the median capacity for this component should be 0.90g as opposed to 1.05g as indicated in LGS-SARA (p. 4-25 of Appendix B). It is further estimated that the mean frequency core melt would increase, approximately, by 20 percent because of the lower median capacity.

(d) SLC Test Tank (S8)

We concur with the recommendations in Reference 1 that the component specific analysis is needed to verify the parameters used in the fragility development.

(e) SLC Tank (S10)

As suggested in Reference 1, the possible failure of the SLC tank due to tearing of the base plate flange near anchor bolts should be checked to verify that it is not the weakest capacity.

4.4 Conclusions

(a) Sources of Conservatism and Unconservatism

Following is a partial and preliminary list of possible sources of conservatism and unconservatism.

(i) Conservatism

- Structural fragility formulation does not have a lower-bound cut-off value. It is believed that below certain acceleration value, an engineered structure or components will not fail.

- The use of low ductility value (2 to 2.5) for flexural mode of failure of shear walls compared to other PRAs (4 to 4.5).

- It appears that median values are generally conservatively estimated.

(ii) Unconservatism

- Omission of explicit consideration of design and construction errors. However, since it is possible errors can lead to either weaker or stronger members this ommission need not always be unconservative.

- Inclusion of duration factor in conjunction with the effective peak acceleration. This is, in part, compensated by the use of low ductility values.

- The modeling uncertainties both due to probabilistic model to determine fragility and original design model, are, generally, underestimated. The effects of increase in these uncertainties are discussed in the Reference 1.

(b) It is concluded that the methodology used in the LGS-SARA is a state-of-the-art approach and this approach, although considered reasonable has not been validated and contains a great deal of uncertainty in itself. The estimated median structural fragility values in LGS-SARA appear reasonable or conservative (except for the case of the reactor enclosure building as discussed in Section 3.0) while the uncertainties are under-estimated in some cases.

(c) In Reference 1, it is indicated that further analyses are required to determine whether the mean frequency of core melt is dominated by contributions from structural failures or electrical component failures. Therefore, the issue of significantly contributing structural components will be discussed at a later date. It should be noted that any significant improvement in risk is not anticipated from any structural fixes.

(d) The methodology used in the LGS-SARA is essentially identical to the one used in the recent PRAs, such as, Zion Probabilistic Safety Study (ZPSS) and Indian Point Probabilistic Safety Study (IPPSS). Following are the major differences between the methodology used in LGS-SARA and ZPSS and IPPSS.

The LGS-SARA includes an explicit factor of safety for earthquake duration (see (a) above).

- The LGS-SARA includes random failure (non-seismic) of components.

- In the ZPSS and IPSS, structural components were found to be dominant seismic contributor to core melt frequency, while in the LGS-SARA electrical components have been found to be dominant seismic contributors to core melt frequency. (see (c) above).

(e) It appears that plant structures and structural components can withstand the earthquake levels well beyond the SSE level. Based on the median acceleration values and associated variability listed in Table 1 in general, no significant probabilities of failure can be identified for these components in the range of the SSE level.

4.5 References

1.0 A Preliminary Review of the Limerick Generating Station Severe Accident Risk Assessment (Draft), Brookhaven National Laboratory, August 15, 1983.

2.0 Review and Evaluation of the Indian Point Probabilistic Safety Study", NUREG/CR-2934, December, 1982.

3.0 Indian Point Probabilistic Safety Study, Power Authority of the State of New York, Consolidated Edison Company of New York, Inc., Spring 1982.

4.0 Letter from Philadelphia Electric Company (PECO) to A. Schwencer of NRC dated August 29, 1983.

5.0 Stevenson, J.D., "Structural Damping Values as a Function of Dynamic Stress," Nuclear Engineering and Design, Volume 60(2) pp. 211-237, September 1980.

Table 1. Significant Structural Fragility Components

No.	Component	Failure Cause or Mode	Median Ground Acceleration Capacity	βR	βR
S_2	Condensate storage tank	Tank-wall rupture	0.24	0.23	0.31
S_3	Reactor internals	Loss of shroud support	0.67	0.28	0.32
S_4	Reactor enclosure and control structure	Shear-wall collapse	1.05	0.31	0.25
S_5	CRD guide tube	Excess bending	1.37	0.28	0.35
S_6	Reactor pressure vessel	Loss of upper support bracket	1.25	0.28	0.22
S_8	SLC test tank	Loss of support	0.71	0.27	0.37
S_{10}	SLC tank	Wall buckle	1.33	0.27	0.19

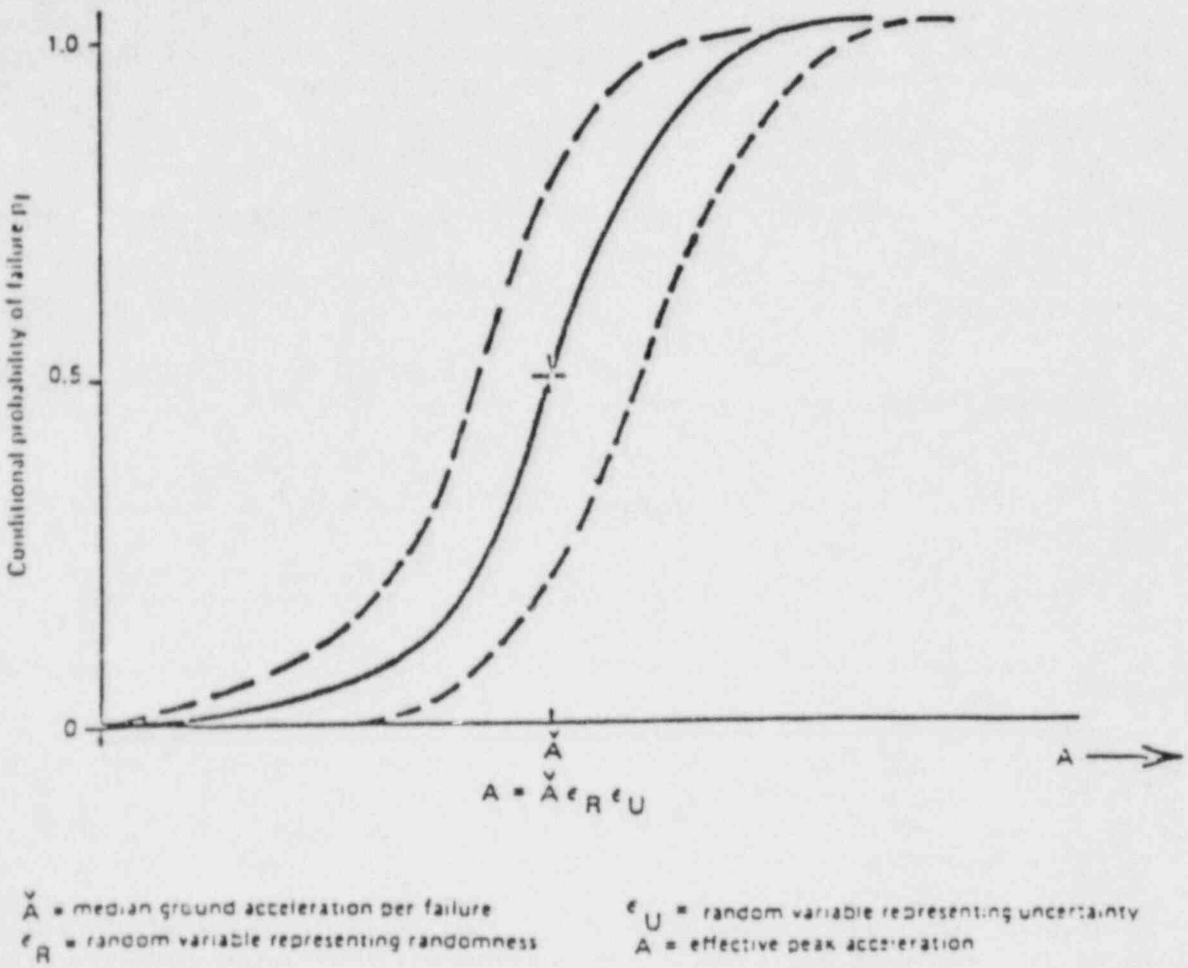

Figure 1.0 Failure probabilities for equipment and structures

5 COMPONENT FRAGILITIES

5.1 Introduction

The evaluation presented here relied upon information provided by the applicant in References 1, 2 and 3, and included the applicable review and evaluation presented in Reference 4 by the Brookhaven National Laboratory (BNL). Our review focused on those portions of Section 3 of Chapter 3, and Sections 1, 2, 3 and 5 of Appendix B of Reference 1 which are pertinent to seismic fragility evaluation of components including piping systems, mechanical equipment and their supports, and the reactor pressure vessel internals.

5.2 Methodology

The seismic risk analysis of LGS-SARA consists of the following four steps.

1. Seismicity: estimation of the occurrence frequencies of ground-motion acceleration.

2. Fragility: estimation of the inability of plant structures and components to withstand various seismically initiated ground accelerations and identification of significant failures.

3. Plant-System Analysis: construction of seismic event tree and fault trees.

4. Accident-Sequence Analysis: quantification of seismic accident sequences.

This evaluation report only addresses the second step regarding the component seismic fragility analysis.

Component seismic fragility is defined as the conditional probability of failure via one of several defined failure modes for a given level of seismic ground-motion acceleration. The objective of a fragility evaluation is to estimate the median effective ground acceleration value for which the seismic response of a given component located at specified point in the structure exceeds the component capacity resulting in its failure via a defined mode. The basis of the method used by the applicant is to first estimate margin to failure or the median factor of safety against failure and its statistical variability under the design basis (generally the safe-shutdown earthquake, SSE). Then the median effective ground acceleration causing failure is estimated by multiplying the SSE (or OBE) acceleration level by this factor.

For equipment and other components, the overall factor of safety is made up of three parts consisting of a capacity factor, an equipment response factor and a structural response factor. The overall factor of safety is then the product of these three factors. The capacity factor is evaluated as the product of the strength factor and the inelastic energy absorption factor (ductility ratio). The strength factor represents the ratio of ultimate strength for the defined failure mode to the stress calculated for SSE. The inelastic energy absorption factor accounts for the fact that equipment is capable of absorbing substantial amounts of energy beyond yield without loss of function. The equipment response factor is the ratio of the realistic equipment response to the equipment response calculated in the design; thus it is the factor of safety inherent in the computation of equipment response. It takes into account earthquake characteristics like the spectral shape, combination of modal response, damping, combination of earthquake components and the method used for the seismic qualification of the component. The structual response factor represents the margin associated with the response characteristics of the structure at the location of the component. It includes structural variables such as spectral shape, damping, modeling and soil-structure interaction. For each factor, a median value was assigned along with a variability due to randomness and uncertainty. When combining the median values, their variabilities are also combined to define the overall safety factor. From this overall safety factor, the effective peak ground acceleration at failure is determined as explained earlier.

The source of information utilized in developing LGS-SARA component seismic fragility and determination of component failure modes will be discussed in detail in Section III, EVALUATION, of this report.

5.3 Evaluation

The objective of this review is to perform an evaluation of the appropriateness of the overall methodology used in the component seismic fragility analyses and a comparison of the LGS-SARA methodology with current state-of-the-art probabilistic Risk Assessment (PRA) technology. Overall, the component seismic fragility analysis of LGS-SARA is found to be reasonably within the state-of-the-art of the current PRA approaches. The staff's evaluation findings and review comments are as follows:

1. In determing the component seismic fragility, the applicant considered both functional failure (i.e., failure to perform its intended function) and structural failure (i.e., failure of the pressure boundary). Because of the variety of equipment to be included in the risk model, the variety of failure modes, and the various sources of fragility information available, it is necessary to treat many groups of equipment generically. Several sources of information are utilized in developing the component seismic fragility in LGS-SARA. These sources include:

    a. Seismic Qualification Review Team (SQRT) Packages

    b. Plant Specific Design Reports, Test Reports

    c. Generic Fragility Test Data from Military Test Programs

    d. Generic Analytical Derivations of Capacity Based on Governing Codes and Standards

    For a piping system, the failure of a single support is very conservatively assumed to result in failure of a piping system. In the analysis of piping systems, pipe support failure rather than plastic collapse of the pressure boundary was found to be the governing case. The inelastic energy absorption (ductility) associated with these failure modes has been considered in determining the seismic fragility of piping system. In References 2 and 3, the applicant indicated that available test data from several sources (References 5, 6, 7, 8, 9, 10, 11 and 12) were utilized in developing piping system fragility. Based on its review of the data base, the applicant concluded that the defined value of plastic collapse used in LGS-SARA together with the available ductility would not result in any serious ovalization which could impair function of the piping system.

As a result of its analysis, the applicant also concluded that the resulting median capacity factor (as defined earlier in Section II of this report) for piping systems is 6.64, a value far greater than any of the structures or components governing seismic risk discussed in Reference 1. Furthermore, the applicant indicated that an additional conservative calculation, without considering the beneficial effect from inelastic energy absorption, was performed to substantiate its conclusion. The resulting peak ground acceleration capacity of piping systems is 1.44g. This is a fragility level significantly greater than many of the other components that played a dominant role in the LGS-SARA.

Based on the information provided by the applicant, the staff determind that the applicant's methodology in determining the seismic fragility for piping systems is found to be reasonably within the state-of-the-art of the current PRA approaches.

With respect to the valve failure modes considered in LGS-SARA, the applicant indicated that the most critical failure mode for a valve is typically the loss of ability to change state due to yielding of the extended operator support. Valve bodies are stronger than the connecting pipe by ASME Code requirements, therefore, valve body rupture is not a credible failure mode. Valves were assumed to fail functionally when a plastic hinge formed in the operator support. An additional failure mode resulting from operator support yielding could be leakage past the stem seals. The applicant further indicated that if malfunction is assumed when the stress in valve operator supports reaches yield strength instead of when a plastic hinge forms, the resulting peak ground acceleration capacity for a valve is 1.87g which is considerably higher than the seismic fragility level of other components that played a dominant role in the LGS-SARA.

Based on the information currently available in References 1 and 2, the staff determined that the applicant's methodology of estimating valve seismic fragility is considered appropriate.

2. With respect to interaction of non-safety related structures or equipment with safety related items, the applicant indicated that a specific system interaction study was not conducted in the Limerick SARA. Major structure/system and component/system interaction potential was, however, addressed. Both safety-related and non safety-related structural failures were analyzed, and the effect of structural failures on safety-related equipment was assessed. A plant walk-through was conducted at Limerick. No

obvious system interactions were observed that were felt to be contributors to seismic risk. In addition, it is noted that the Limerick design requires that non-safety related components which are located in the vicinity of safety-related items are either analytically checked to confirm their integrity against collapse when subjected to seismic loading from the SSE or are separated from seismic Category I equipment by a barrier. However, it is possible that the responses and capacities, i.e., the factors of safety to withstand seismic ground motion acceleration above SSE, for some non-safety related items are different from the nearby safety-related items. Because the walk-through was conducted before the completion of construction, a confirmatory assessment and walk-through should be conducted after construction of the plant is completed to locate non-safety related components which could fail, fall and impact safety-related items. Consideration should be given to the possible effects of actual response and capacity characteristics to determine whether the non-safety related items are weaker than the nearby safety-related items.

3. In Reference 1, the applicant indicated that stresses resulting from seismic and normal loadings are utilized in determining the component seismic fragility. LOCA and other dynamic loading combined with seismic events is considered too low a probability combination to be included in the development of seismic fragility. In Reference 2, the applicant also stated that pipe support failure were found to be the governing case for piping systems in LGS-SARA. For the design of pipe supports at Limerick, the stress produced by the seismic anchor point motion of piping in the supports is considered as primary stress which is utilized in the fragility calculation. Furthermore, when inelastic deformation of the structure resulting from large structural displacements beyond the elastic limits is estimated (engineering judgement not analytical evaluation) to be sufficient to potentially damage the equipment attached to the structure, systems connected to the structures are considered to fail. Therefore, the applicant concluded that the differential movement of structures are implicitly considered in the development of fragilities for piping systems.

   Based on the information provided by the applicant, the staff determined that in general, the applicant's procedures are considered appropriate. However, for the case of potential impact between the reactor building and containment, the staff agreed with the BNL review comment as addressed in Section 2.1.3.5 of Reference 4. All the safety-related piping which connects both buildings should be systematically reviewed to verify that sufficient flexibility is provided to accommodate relative displacement between the structures.

   With respect to potential impact due to tilting of structures, the applicant indicated that Limerick structures are formed on component rock and tilting of such structure is not a credible failure mode. Therefore, failure of piping and equipment due to tilting of the structure is not a credible failure mode. In addition, the applicant stated that buried pipe at rock sites is not subjected to excessive strains unless there is excessive block motion of the rock. Large block motion is not anticipated to occur at earthquake levels that cause failures of major structures and equipment. Therefore, buried pipes are not considered to be a governing element in the accident sequences. However, it is noted that a portion of the Limerick emergency service water cooling line is buried in backfill material of which the soil amplification factor could be greatly different from the rock. The applicant should verify that the potential failure of the buried pipe due to soil amplification is not credible. Evaluation of the appropriateness of the soil amplification factor is being reviewed by the Structural and Geotechnical Engineering Branch (SGEB).

4. Three of the significant earthquake-induced component failures of LGS-SARA are associated with the reactor pressure vessel (RPV). These include reactor internals, RPV and the CRD guide tubes. The RPV is supported by a base skirt and an upper lateral supports. The upper portion of the RPV is supported by a lateral stabilizer which spans the gap between the RPV and reactor shield wall. The reactor shield wall in turn, is anchored to the containment wall by a steel seismic truss. In Reference 1, the applicant stated that the RPV support reactions are predominantly a function of the dynamic characteristics of concrete support structure rather then dynamic characteristics of the RPV itself. In the development of the median capacity factor for the reactor internals, RPV, and the CRD guide tubes, it was assumed that the containment structure had an effective damping value of 10% which differs from the 5% damping used in the original design analysis addressed in LGS-FSAR Section 3.8. The appropriateness of the applicant's analyses for concrete support structure and their impacts on final results, i.e., the validities and uncertainties associated with these components fragility analysis, are being reviewed and addressed by the SGEB.

Based on its assessment, the applicant has identified 17 key components and structures whose failures affected the dominant sequences leading to core melt. A copy of the fragility calculations for these 17 significant components listed in Table 3-1 of Reference 1 were reviewed by the NRC consultant, Jack R. Benjamin and Associates, Inc. (JBA). The detailed results of this review are addressed in Section 2.1.3.7 of Reference 4. The significant earthquake-induced failures pertinent to the Mechanical Engineering Branch review included failure of the reactor vessel internals shroud support, failure of the RPV upper support bracket, failure of CRD guide tubes, failure of hydraulic control unit, failure of the nitrogen accumulator anchor bolt and failure of the diesel generator heat and vent system support. The staff agrees with the review comments addressed in Section 2.1.3.7 of Reference 3. The applicant's methodology of evaluating the seismic fragilities for reactor internals, RPV and the CRD guide tubes is appropriate except the item addressed in Section III.4 above. WIth respect to hydraulic control unit, nitrogen accumulator and diesel generator heat and vent system, in general, the applicant's results, i.e., the median capacity factors and the uncertainties are conservative. However, for these components, the analyses are based on either generic capacities or Susquehanna Nuclear Power Plant Tests and fragility calculations. Since they are important to the final risk, specific calculations based on the characteristics of these components for Limerick plant should be performed. Alternatively, the applicant should provide information to justify the validity of its data base including the consideration of the possible differences of component capacities and responses due to the different foundation conditions, installation and construction of the component.

5. With respect to design and construction errors, the applicant indicated that an inadequate data base exists upon which to determine explicitly the contributions of design and construction errors to most Limerick structures and equipment seismic capacities. Minor design and construction errors are accounted for in the variabilities associated with the various modes of failure investigated. However, only gross errors can influence the seismic risk results for a nuclear power plant. In general, for a plant as new as Limerick with current design and QA procedures, the possibility is considered remote that major design and construction errors exist which can significantly affect the seismic capacity of a component. Nevertheless, there is a possibility that unidentified design and construction errors may exist which can affect the seismic capacity. Since the LGS-SARA analysis does not include a comprehensive consideration of design and construction errors and hence, may be biased (note that errors may be conservative or unconservative), the results are useful only in making relative comparisons.

5.4 Conclusion

Based on its review of the information in Reference 1, 2, 3 and 4, the staff concluded that the methodology used in the LGS-SARA for component seismic fragility analysis is appropriate and adequate to obtain a relative measure of the seismic capability of the Limerick plant. As addressed in Reference 4, the mean frequency of seismically induced core melt in LGS-SARA is dominated primarily by electrical components. This differs from the Indian Point Probabilistic Safety Study (IPPSS) and the Zion Probabilistic Safety Study (ZPSS), which the non-electrical components and structures controlled the results. As addressed in Section III, Evaluation, of this report, the procedures used in LGS-SARA in determining the component seismic fragility are based on limited available test data and rely heavily on engineering judgement. The analysis does not include a comprehensive consideration of design and construction errors and, hence, may be biased (note that errors may be either conservative or unconservative). The specific issus and comments raised in Section III of this report need to be resolved before judging the impact on final results, i.e., the validity and uncertainty associated with numerical estimation of component seismic fragility. Nevertheless, the results from the LGS-SARA are useful in obtaining a relative measure of seismic capability of components for Limerick plant and should not be viewed in an absolute sense. The staff has been able to conclude that the component seismic fragility analysis of LGS-SARA is reasonably within the state-of-the-art of the current PRA approaches.

5.5 References

1. Report, "Severe Accident Risk Assessment, Limerick Generating Station," Philadelphia Electric Company, dated April 1982.

2. Letter, John S. Kemper to A. Schwencer, "Response to NRC Questions on the Severe Accident Risk Assessment," Philadelphia Electric Company, dated August 24, 1983.

3. Letter, John S. Kemper to A. Schwencer, "Response to NRC Questions on the Severe Accident Risk Assessment," Philadelphia Electric Company, dated August 29, 1983.

4. Report, "A Preliminary Review of the Limerick Generating Station, Severe Accident Risk Assessment," Engineering and Risk Assessment Division, Department of Nuclear Energy, Brookhaven National Laboratory, dated August 15, 1983.

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6 EQUIPMENT FRAGILITIES

6.1 Scope

The review comments and evaluation presented here are based mainly on the review of Chapter 3 and Appendix B of Limerick Generating Station Severe Accident Risk Assessment (LGS-SARA), (Reference (1)), in relation to equipment fragility. The purpose of the review is to present the staff comments regarding the adequacy of the approach taken by the applicant as well as the findings of the study. During the review process, frequent interactions have been maintained between the staff and the Brookhaven National Laboratory (BNL), the staff's consultant on this review task. As a result, BNL's evaluation forms the main body of the review comments.

6.2 Methodology of SARA

Fragility descriptions for each of those equipment items participating in the accident sequences were developed utilizing available information from Utility, Architect engineer, the NSSS vendor, and other sources of fragility information.

Fragility levels are expressed as frequencies of failure vs effective peak ground acceleration. The procedure used in deriving fragility descriptions is similar to that used for structural fragility descriptions, wherein, median factors of safety and variability are first developed for equipment capacity and equipment response. These two factors, along with the median factor of safety on structural response, are then multiplied together to obtain an overll median factor of safety for the equipment items.

The logarithmic standard deviations associated with the above individual factors are combined by Root-Sum-Square (SRSS) method to establish an overall variability on the equipment fragility. The logarithmic standard deviations are further divided into variabilities due to randomness and uncertainty.

The median overall factor of safety obtained is then multiplied by the reference earthquake peak ground acceleration to arrive at the equipment capacity in terms of peak ground acceleration.

The SSE is generally used as the reference earthquake.

Several sources of information were used to derive plant specific and generic fragilities for equipment. These sources include (a) Final Safety Analysis Report (FSAR), (b) Seismic Qualification Review Team (SQRT) submittals for the Susquehanna Steam Electric Plant, (C) United States Corps of Engineers Shock Test Reports, and (d) Other Probabilistic Risk Assessment (PRA) Reports.

Susquehanna is a very similar plant designed and constructed by the same Architect Engineer (Bechtel Corporation) and same NSSS Vendor (General Electric Co.) as Limerick. Consequently, advantage was taken of previous fragility description derivations for Susquehanna equipment where applicable. Many of those derivations were based on information contained in Susquehanna SQRT submittals, in which summaries of analysis or test methods and results are available. Analysis results can be used directly to develop fragility descriptions. Test results are used in conjunction with generic fragility test data to develop approximate fragility levels. In some cases, capacities are developed from summaries of design reports contained in the FSAR.

Some generic fragility test data were utilized in the derivation of fragility descriptions. Fragility tests and severe shock environment tests have been conducted for off-the-shelf type equipment similar to electro-mechanical, electrical and control equipment installed in nuclear power plants. The results of some 60 test programs are summarized by U.S. Corps of Engineers. Information from these shock test reports are used in deriving generic capacities of equipment where plant specific information was not readily available or could not be extrapolated to a failure level.

Because of the variety of equipment to be included in the risk model, the variety of failure modes, and the various sources of fragility information available, it is necessary to divide the equipment items into distinct groups. The selected major categories of equipment are:

1. Plant specific equipment whose fragility descriptions are based on structural failure and for which summaries of design reports are reviewed.

2. Plant specific equipment whose fragility descriptions are based on functional limits and for which summaries of design reports were reviewed.

3. Equipment for which generic structural capacities can be derived from knowledge of the design specifications and the strength factors of safety inherent in the governing codes and standards.

4. Equipment for which generic structural and functional capacities can be derived from fragility test data, military shock test data, seismic qualification test reports or other generic tests.

5. Valves for which generic structural and functional capacities can be derived from sampling of capacities of several valves qualified for Susquehanna or other nuclear power plants, and from shock tests of piping systems containing valves.

The equipment response factor, as mentioned previously, is a measure of the conservatism or nonconservatism and the associated variability in determining the seismic response of equipment. Because of the variety of methods used in qualifying equipment for seismic service, the response factor derivations are further grouped into the following several generic qualification categories:

1. Equipment qualified by dynamic analysis
2. Equipment qualified by static analysis
3. Equipment qualified by test

Depending upon the specific categories of equipment and its qualification method, the pertinent variables that affect the computed response and its dispersion may consist of some of the following parameters:

1. Spectral shape
2. Static coefficient used vs spectral acceleration
3. Modeling
4. Damping
5. Boundary conditions in the test vs installation
6. Equipment fundamental frequency
7. Combination of modal responses
8. Combination of earthquake components
9. Test method (sine beats, sine sweep, complex wave form, etc.)
10. Multi-axial coupling and directional component

Equipment capacity factors and their variabilities, on the other hand, can be derived for each of the equipment categories by considering the affecting parameters, such as strength factors based on static strength, and/or ductility factor based on inelastic energy absorption much in the same manner as for structures. The capacity factor is then the product of the strength and ductility factors, as applicable.

Based on the above mentioned methodology, the applicant has compiled an information summary (see LGS-SARA Appendix B Table 5-5) of equipment capacity, equipment response, and structural response factors, their logarithmic standard deviations and the median ground acceleration capacities for equipment items identified as important to the seismic risk study. Systems for which the information is included consist of Reactor Core Isolation Cooling (RCIC) Systems, High Pressure Coolant Injection (HPCI) System, Core Spray Systems, Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI) Systems, Automatic Depressurization System (ADS), Electrical Power System, Emergency Service Water System, Standby Liquid Control System, Reactor System, and Scram System.

In the event/fault tree modeling for LGS-SARA, individual fault trees for the mitigating systems represented in the seismic event tree were modified by incorporating seismic component failures into the system fault trees developed for the analysis of internal events and modularizing the system fault trees until each system fault tree is represented by a non-seismic failure, a seismic failure or failures, or a combination of seismic and non-seismic failures. Each individual fault tree was further reduced by neglecting seismic failures with very low probabilities of occurrence and hence negligible contributions to core-melt frequency. Through this process it was possible to identify 17 significant seismic failures, each of which has a median ground acceleration capacity of less than or equal to 1.56 g. In LGS-SARA Table 3-1, these 17 components are listed together with the corresponding failure mode, and the median ground acceleration capacity and its variabilities.

6.3 Evaluation Finding

As stated in the BNL evaluation report (Reference (2)), Jack R. Benjamin and Associates, Inc. (JBA) was retained by BNL to perform a preliminary review of the LGS-SARA for the effects of seismic events. JBA has performed similar reviews for the Indian Point Probabilistic Safety Study (IPPSS) and the Zion Probabilistic Safety Study (ZPSS).

It is noted that the hazard and fragility calculations for LGS-SARA were performed by the same engineers and were based on the identical methodologies used for the IPPSS and ZPSS, and therefore many of the issues and concerns generic to all sites and plants already have been discussed and evaluated (Reference (3)). As a result, the current JBA review focused primarily on critical areas which may impact the results and documented the important concerns applicable to the Limerick plant.

As part of the review, a meeting was held at the Structural Mechanics Associates (SMA) office in Newport Beach, California, on July 8, 1983, with representatives from NRC, SMA, JBA, Nuclear Utility Service (NUS), and Dames and Moore participating to discuss issues raised to date concerning the LGS-SARA and to direct the review effort on the critical components and issues. Subsequent to this meeting, a tour of the Limerick plant was conducted on July 15, 1983, with representatives from NRC, JBA, and NUS participating to review the installation of the above mentioned significant components as well as other equipment items of interest whose fragilities were reviewed in relationships to their potential impact on the mean frequency of core melt.

For example, the median capacity of the batteries and racks is reported to be as high as 2.56 g and thus was not included in the sequences. This component was inspected during the plant tour, and its capacity value is judged to be reasonable.

Based on the staff evaluation of LGS-SARA, the applicant's submittal of August 24, 1983 responding to staff questions, and the recommendation provided by BNL (Reference (2)), the following comments on the LGS-SARA in regard to equipment in general as well as significant components in particular are presented for applicant's consideration.

1. Impact between the reactor building and containment might cause high frequency motion to safety-related electrical equipment items located in the reactor building. The capacities of these equipment items range between 1.46 g and 1.56 g. This is considerably higher than the motion level of 0.45 g at which impact may occur; hence, these capacities may, in reality, be less than the above estimated values. Additional study by the applicant may determine the actual effect of the impact and consequently the realistic reduction of the capacity of the electrical equipment.

2. The staff was concerned about the effect of a potential, excessive leakage of mechanical components, such as main steam isolation valves (MSIV's), on the integrity of pressure boundary as well as other related systems. In the August 24, 1983 submittal the applicant states that for MSIV, which is normally open and would have to change state after a seismic event, the critical failure mode is more likely to be failure-to-close than leakage once the valve closes. The fragility description for the MSIV was developed for a failure mode to close, which is judged to be the lowest capacity and the governing mode of failure. The staff has found this response acceptable.

3. The staff was concerned about the fragility decription of purge and vent valves which were not specifically addressed in LGS-SARA.

The applicant, in his August 24, 1983 submittal, states that valves in Limerick have, in general, an estimated peak ground acceleration capacity of around 1.87 g which is considerably higher than any of the fragilities that played a dominant role in the SARA. He further states that for purge and vent valves, similar capacities are anticipated. In addition, these purge and vent valves are normally closed (typically open only 1 percent of the time) and significant leakage would require even higher accelerations. The staff found the above response acceptable and determine that purge and vent valves play an insignificant role in LGS-SARA.

4. In regard to relay chatter, although reset of the system may be readily possible at the control room under certain circumstances, there are those relay trips which may require resetting at local panels and would cause high failure probability of the operator to reset; failure to do so would result in the equivalent of a relay failure. Furthermore, there is the underlying question that, in view of the different modes of relay trips, the operator may be presented with a scenario for which he has not been trained and for which no written procedure is available for guidance, what would be the probability that he will perform adequately to reset the relays. It is felt that LGS-SARA should include additional analysis on the relay chatter and address its impact upon various systems. Failure of human action required to reset under stressed condition, and hence leading to relay failure, should be considered.

5. For Offsite Power (500/230-KV Switchyard) (S_1) in LGS-SARA Table 3-1, the fragility description is based on the failure of porcelain ceramic insulators. Based on historic data, the capacity is estimated at 0.2 g and appears to be reasonable.

6. For 440-V Bus/SG Breakers (S_{11}), power circuit failure was identified as the failure mode. The capacity was developed based on test data from the Susquehanna SQRT submittals. The median capacity from Susquehanna was scaled by the ratio of the two SSE peak ground acceleration values.

   Although the fragility parameter values (acceleration capacity is 1.46 g, standard deviations for randomness and uncertainty are 0.38 and 0.44 respectively) appear reasonable, it is not apparent whether the calculation by scaling has considered such important factors as differences in foundation condition, and, hence, the response of the reactor building, and the locations of the corresponding components in the two plants.

   Since this component is a significant seismic contributor to the mean frequency of core melt, it is felt that a specific analysis should be conducted for this component.

7. For 440-V Bus Transformer Breaker (S_{12}), 125/250-V DC Bus (S_{13}), and 4 KV Bus/SG (S_{14}), the failure modes were identified to be loss of function, loss of function, and breaker trip respectively. The capacities for these three components are the same (acceleration capacity is 1.49 g, standard deviations for randomness and uncertainty are 0.36 and 0.43 respectively) and are based on the fragility analysis of the diesel generator circuit breakers which, in turn, are based on the analysis of

test data for the Susquehanna plant. Same comments as given for S_{11} are, therefore, also applicable.

In summary, the fragility description for these three components appears reasonable. However, because they are significant seismic contributors to the mean frequency of core melt, it is felt that a specific component analysis should be conducted for each.

8. For Diesel Generator Circuit Breakers (S_{15}), the failure mode identified is loss of function. The fragility parameters (acceleration capacity is 1.56 g, standard deviations for randomness and uncertainty are 0.32 and 0.41 respectively) appear reasonable. However, for the same reasons as stated for S_{11}, S_{12}, S_{13}, and S_{14}, it is felt that a specific analysis should be conducted for this component.

6.4 Conclusion

Based on our review it is felt that the methodology and approach used in LGS-SARA in developing equipment fragilities are generally acceptable and the fragility descriptions presented appear reasonable. However, more supporting information needs to be furnished in LGS-SARA to resolve the comments presented in Section 3.0 before final staff judgement of the adequacy of the methodology and the results of the study can be made.

It appears that LGS-SARA differs from the IPPSS and ZPSS in that the seismic contribution to the mean frequency of core melt is dominated primarily by five electrical components in series (see Item Nos. 6, 7 and 8, Section 3.0), which have nearly the same median capacities. In contrast, nonelectrical components and structures controlled the results of the IPPSS and ZPSS. The capacities for the LGS-SARA electrical components are derived based on generic tests and are not component specific. This approach is reasonable as long as the components do not control the final results. Since the electrical components are significant seismic contributors to the PRA, a more detail analysis regarding equipment fragility should be conducted, as recommended in Section 3.0.

6.5 References

(1) Severe Accident Risk Assessment, Limerick Generating Station, Philadelphi Electric Company, Report No. 4161, dated April, 1983.

(2) Azarm, M. A., et al., "A Preliminary Review of the Limerick Generating Station Severe Accident Risk Assessment," Volume I: Core Melt Frequency, Brookhaven National Laboratory, draft NUREG/CR Report, dated August 15, 1983.

(3) Kolb, G. J., et al., "Review and Evaluation of the Indian Point Probabilistic Safety Study," Prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-2934, December, 1982.

7 RECOMMENDATIONS AND SUMMARY

Our review and that of BNL have indicated areas of the earthquake related portion of the LGS-SARA that could be improved by additional clarification and sensitivity studies. These recommendations, outlined in each of the previous sections on seismic hazard and structural, mechanical, component and equipment fragilities, address a wide range of specific seismological and engineering topics.

While addressing specific issues, such as the postulated error in a Boolean expression pointed out by BNL, would improve the LGS-SARA, there are several fundamental shortcomings of the seismic event PRA which are inherent in performing a seismic PRA and it is beyond our means to adequately address them. First and foremost is the inadequacy of the existing historical and instrumental seismic record (two to three hundred years). PRAs try to utilize this record to draw inferences on earthquakes that appear to have mean return periods on the order of tens and hundreds of thousands and perhaps even millions of years. This extrapolation to provide rigourous numerical estimates must be viewed as highly speculative, particularly since we lack a fundamental understanding of the causative nature of the earthquake potential in the eastern U.S. Attempts to deal with the problem lead to the observation that most of the calculated uncertainty in seismic event PRAs, such as LGS-SARA, is related to uncertainty in the seismic hazard. A second problem relates to the fact that the characterization of fragility is based on little data and a great deal of engineering judgement. Finally there are some aspects of the problem where useful and comprehensive models incorporating engineering judgement have not even been proposed. In the LGS-SARA, design and construction errors fall into this category. As a result, there exists a significant potential for systematic bias that cannot be simply accounted for. However when making relative comparisons, that is ratios, where such biases may be common to the entities being compared (e.g determining which are the major contributors to seismic risk), then errors resulting from them tend to be minimized. We therefore agree with our consultant (BNL) who states that "the results from the LGS-SARA are useful in a relative sense and should not be viewed as absolute numbers".

It is our judgement that reliance upon simple point estimate such as means or medians to characterize actual risk may be premature. However there has been an extensive effort to define the uncertainty. The wide bands of uncertainty presented in relation to the seismic elements of the LGS-SARA can be thought as representing a large part, but not all, of the actual uncertainties. They may be used to gain insight as to the range of the actual risk associated with seismic initiating events at Limerick. We do not mean to imply that higher risk estimates (e.g. 95th percentile) are more appropriate than the median, mean or lower (5th percentile) estimates. Indeed the most significant earthquake damage anywhere within the vicinity of the Limerick Site, in the two to three hundred years during which we have records, are fallen chimneys 50 kilometers away during an earthquake at Wilmington, Delaware in 1871 whose magnitude can be estimated to have been less than 5.0. We certainly cannot exclude from the range of reasonable assumptions the judgement that there essentially is no risk to the public resulting from earthquake-induced damage at the seismically-engineered nuclear power plant at Limerick during its operating life.

The nature of seismic PRAs such as the LGS-SARA requires us to look at the behavior and fragility of plants at ground motion levels well beyond the SSE as evidenced by Table 3.1 (Significant Earthquake Induced Failures) of the LGS-SARA. Even though some of these ground motion levels may appear extremely high for such a seismically quiet site as Limerick, they do provide us with insight as to the seismic capacity of the plant. For example, the applicant in response to NRC questions, estimates that the reactor and control buildings shear walls have a 95% confidence of less than a 5% failure fracture at approximately twice the SSE. Although such conclusions are based upon the generalized assumptions needed to carry out the LGS-SARA, it is our judgement based on past experience that a detailed seismic margins analysis would support the conclusion that the Limerick Nuclear Power Plant can withstand postulated earthquake ground motion well beyond that defined by the SSE.

Finally a seismic PRA affords an opportunity to examine postulated accident chains and sequences that could lead to serious damage and result in radioactive release. Our review of the LGS-SARA indicates that there are no meaningful outliers in Table 3.1 of the SARA (Significant Earthquake Induced Failures) such that simple modification to any of these structures, components and equipment would result in a significant reduction in risk to the public.

APPENDIX D

SEISMIC HAZARD CHARACTERIZATION OF THE EASTERN UNITED STATES

Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites

This appendix discusses the uncertainty associated with calculations of the seismic hazard used in the Limerick Severe Accident Assessment (SARA).

The staff addressed this issue in the Limerick Final Environmental Statement (FES) as follows.

> Severe earthquakes are one cause of accidents. Uncertainties in the estimates of probabilities of severe earthquake induced core melt sequences are judged to be very large because of (1) the relatively sparse data base on severe earthquakes in the eastern U.S. and (2) the unavailability of an acceptably precise and definite procedure to quantify seismically induced accident sequences. In LGS-SARA, the spectrum of probabilities of seismically induced core melt sequences varied over a wide range (several orders) or magnitudes. However, the mean (point or best estimate) probabilities of seismically induced core melt accident sequences used in the staff analysis (which essentially came from LGS-SARA) are within the range of probabilities developed in LGS-SARA, and are within a factor of about 6 of the upper end of the spectrum of probabilities in LGS-SARA. Thus, the point estimates of seismic probabilities used to evaluate risks are more representative of Limerick than WASH-1400 values, and consider the applicant's estimate of the range of seismic frequency uncertainty. The staff has concluded that the high and low values of the range should not be characterized as 95% and 5% limits, but rather as a representative range of the seismic sequence frequencies, which incorporates a large part (but not necessarily all) of the uncertainties with such events. This statement reflects the staff's view that the rigorous definition of seismic hazard and its uncertainty at low probabilities is beyond the state-of-the-art at this time and should be recognized as such. Different studies would not necessarily yield equivalent results. For example an interim report to be published "Seismic Hazard Characterization of the Eastern U.S." of an ongoing study being carried out by Lawrence Livermore National Laboratory (LLNL) for the NRC shows seismic hazard calculations for the Limerick site which overlap, but are not necessarily coincident with, the range of seismic hazard assumed in LGS-SARA.

> The median (50%) hazard calculated in the interim LLNL report is within, but near the high end of, the range of hazard curves utilized in LGS-SARA. Additional studies of seismic hazard in the eastern U.S. are being carried out by such groups as the Electric Power Research Institute. Given the highly judgmental nature of seismic hazard calculations, there is not reason to believe that these studies or the final LLNL report would not show differences in estimated seismic hazard and uncertainty between themselves and the LGS-SARA, particularly at the low probabilities being calculated for Limerick.

In the FES, as noted above, the staff discussed the interim results from the Lawrence Livermore National Laboratory Seismic Hazard Characterization Program. These results have been published in the report entitled "Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites" NUREG/CR-3756, April 1984. These results further emphasize the uncertainty associated with calculations of seismic hazard used in the LGS-SARA. They diverge from the LGS-SARA results particularly at frequencies less than 10^{-5} per year. The staff does not anticipate better agreement between commercial PRAs and the LLNL calculations at other sites.

At this time the staff does not necessarily believe that one is wrong and the other is right. The NRC staff is attempting to evaluate and determine to what extent this divergence is the result of inherent uncertainties in state-of-the-art hazard estimates or systematic errors in input assumptions. Substantial advances in the understanding of earthquake causality and ground motion may be needed to significantly improve the picture. As a result the staff would like to reemphasize the oft-stated preference (see for example NUREG-1050) of not placing absolute confidence in bottom line numbers produced in seismic PRAs. Central estimates (means or medians) are vulnerable to the highly judgmental choice of input parameters. In addition, uncertainty bands cannot be rigorously defined. To the extent that the generation of such numbers is desired, the following (in whole or in part) may be used to put such calculations in the proper context.

1. Central estimates should be deemphasized and greater weight given to uncertainty bands stressing, however, that these bands may not be accurately defined and should be viewed as "representative ranges."

2. Displays of any calculations should avoid more than one significant figure since the uncertainty is certainly several orders of magnitude.

3. Calculations may be put in context through the use of qualitative descriptions and qualifiers (see for example the above statements from the Limerick FES).

4. One future approach to dealing with the uncertainty may be to define the hazard levels (or range) above which qualitative conclusions with respect to a particular nuclear power plant would change. Qualitative and/or quantitative discussions could then be used to describe why reasonable estimates of the hazard would lie above or below that level (or range).

5. Further study could be useful to answer the question as to whether the presence of a nuclear power plant significantly increases the risk to the public during a severe earthquake (the kind needed to cause a core melt). If the argument can be made that this incremental risk is negligible then the need for reliable estimates of rare earthquake occurrence may be greatly diminished for the purpose of environmental statements. An initial examination of this issue indicates that this statement cannot be made at this time.

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In recognition of the high population density around the Limerick Generating Station site and the proposed power level, the Philadelphia Electric Company, in response to NRC staff requests, conducted and submitted between March 1981 and November 1983 a probabilistic risk assessment (PRA) on internal event contributors and a severe accident risk assessment on external event contributors to asess risks posed by operation of the plant. The applicant has developed perspectives using PRA models on the risk profile of the Limerick plant and has altered the plant design to reduce accident vulnerabilities identified in these PRAs. The staff's review of the Limerick PRA has particularly emphasized the dominant accident seqences and the resulting insights into demonstration of compliance with regulatory requirements, unique design features and major plant vulnerabilities to assess the need for any additional measures to further improve the safety of the LGS. The staff's review insights and PRA safety review conclusions are presented in this report.

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FOR THE LIMERICK GENERATING STATION

AUGUST 1984