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H. LUDEWIG, J. W. YANG, AND W. T. PRATT

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ACCIDENT ANALYSIS GROUP

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ABSTRACT

The objective of this report is to calculate a range of source terms that will be used as input to the site consequence analysis to be performed by the NRC staff as part of the Draft Environmental Statement (DES) and the Final Environmental Statement (FES) for the Limerick Generating Station (LGS). At the direction of NRC staff, a "base case" approach was used in the development of these source terms. This approach relies on the Reactor Safety Study methods (or prescriptions) to determine the release of fission products from the damaged fuel, primary system hold-up, suppression pool scrubbing and transportation of the fission products in containment. The report utilizes information in the LGS-Probabilistic Risk Assessment (internal initiated events) and in the LGS-Severe Accident Risk Assessment (external initiated events) and in BNL reviews of these documents to assess the probabilities of potential containment building failure modes and release paths. However, the use of the "base case" approach necessitated extensive reanalysis, using the MARCH/CORRAL system of codes, to develop appropriate source terms. In all, twenty-seven source terms have been developed to represent all potential failure modes and release paths in the LGS. Each of these source terms provide the fraction of fission product species released to the atmosphere, the characteristics of the release and the frequency of occurrence of the release.

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The work is being performed for the Division of Systems Integration (DSI) at the U. S. Nuclear Regulatory Commission (NRC). J. Carter is the NRC technical monitor for this project, however, J. Meyer had significant input to this report prior to his leaving DSI. J. Rosenthal is the NRC section leader for this project.

We would like to express our appreciation to Theresa Rowland for her excellent typing and assembling of this report.

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1.0 INTRODUCTION

This section gives the background to our assessment, indicates the objectives of our work, and describes the way in which the report is organized.

1.1 Background

In March 1981 a Probabilistic Risk Assessment[1] for the Limerick Generating Station (PRA-LGS) was submitted to NRC. The LGS-PRA considered accident sequences initiated only by internal events. In February 1983 Brookhaven National Laboratory (BNL) issued a detailed review[2] of the LGS-PRA. In April 1983 contractors (NUS) to the Philadelphia Electric Company (PECo) completed a study[3] which included an evaluation of risk due to seismic initiating events and to fires that might be initiated within the plant. This study, the Severe Accident Risk Assessment for the Limerick Generating Station (LGS-SARA), used generic accident classes developed in the LGS-PRA whenever possible to also represent accident sequences initiated by external events. However, because of the unique characteristics of some of the seismically initiated sequences, additional accident classes were developed in the LGS-SARA. In addition, the LGS-SARA included a revised analysis of the off-site consequence analysis using the CRAC2[4] computer code.

In June 1983 NRC requested that BNL undertake a preliminary, short-term review of the LGS-SARA. The review will be contained in a two-volume report.[5] Volume I reports on the review of seismic and fire methodologies as they relate to the determination of the core melt frequency. Volume II will report on the BNL review of core melt phenomenology, fission product behavior, and off-site consequences will be issued at a later date.

In addition to a review of the LGS-SARA, the NRC also requested that BNL provide input to the Draft Environmental Statement (DES) for the LGS and a draft version[6] of this report was the product of that effort. Specifically, NRC requested that BNL generate a complete set of source terms for core meltdown accidents in the LGS initiated by internal and external events. The DES source terms developed in the draft report were based on information in the LGS-PRA, LGS-SARA and on BNL reviews of these documents. These DES source terms were sent via an NRC memorandum[7] to the Accident Evaluation Branch (AEB) who used them to perform the LGS site consequence analysis, which was published[8] in the LGS-DES. The only input BNL had to the LGS-DES was the source terms described in Section 5 of this report.

Following the publication of the LGS-DES, BNL was requested to reanalyze the Class IV sequences. This reanalysis was to take into account improved thermal/hydraulic modeling and the presence of the reactor building. The recalculated Class IV source terms are included in the present version of this report for input to the LGS Final Environmental Statement (FES).

The approach taken to the development of the source terms in the LGS-PRA was to utilize the methods (or prescriptions) used in the Reactor Safety Study.[9] Consequently, the release of fission products from the damaged fuel was assumed in the LGS-PRA to follow the Gap, Melt, Oxidation and Vaporization release phases described in Reference [9] and in Section 4 of this report. However, the methods used in the LGS-PRA did differ from the RSS in a number of respects, namely:

- temporary hold-up of fission products released during the gap and melt release phases in the primary system until after vessel failure

- a decontamination factor of 10 was used for a saturated suppression pool (no decontamination was assumed in the RSS for a saturated pool).

The approach taken in the LGS-SARA to develop source terms differed even further from the methods used in the RSS. The release of fission products (FP) during the in-vessel melt release phase was assumed to follow the trends reported in NUREG-0772.[10] Thus, significantly higher fractions of all aerosols are released ex-vessel in the LGS-PRA than in the LGS-SARA.

There has been significant research activity in this area since the publication of the RSS in 1975. A basis for estimating FP behavior was published[10] in 1981 by RES/NRC and has been used in the LGS-SARA. In addition, updated fission product source term methods are currently being developed[11] and are receiving extensive peer review. At this stage, BNL is unable to confirm the validity of the changes made in the LGS-PRA and LGS-SARA relative to the RSS approach. Hence, at the direction of NRC staff, the approach taken in this report to the development of the source terms is to follow the RSS prescriptions regarding the release of fission products from the damaged fuel, primary system hold-up, suppression pool scrubbing and transportation of the fission products in containment. The use of this approach by BNL staff in this particular application does not constitute technical endorsement of the assumptions, data or techniques that are associated with this approach. The aim is simply to calculate a consistent set of source terms that are applicable to both internally and externally initiated accidents based on RSS prescriptions. The use of the RSS based source terms is part of the "base-case" approach described in Reference [12] and in Section 4 of this report. Reference [12] also gives the justification for such an approach.

1.2 Objectives and Scope of the Analysis

The objective of this report is to provide a complete set of source terms for core meltdown accidents in the LGS initiated by internal and external events. Each of these source terms will provide the fraction of fission product species released to the atmosphere, the characteristics of the release and the frequency of occurrence of the release. These source terms will be used as input to the site consequences analysis to be performed by the NRC staff as part of the LGS-DES and LGS-FES.

The classification of the core-melt accident sequences in this report has been made consistent with References [1]-[3]. The LGS-PRA[1] and the BNL review[2] were used to classify accidents initiated by internal events. The LGS-SARA[3] was used to classify externally initiated accidents. The probabilities of the internally initiated accident classes were based on the BNL review[2] of the LGS-PRA. The LGS-SARA[3] was used as the basis for determining the probabilities of external events. The only changes that were made to the frequencies and accident classifications reported in the LGS-SARA relates to the influence of seismic events on evacuation. In the LGS-SARA it was assumed that evacuation would only be influenced by effective peak accelerations in excess of 0.61g (refer to Section 10.1.6.5 of Reference 3). This was considered inappropriate by BNL Contractor J. Reed (refer to the BNL review of the LGS-SARA, Reference 5) and accelerations in excess of 0.4g were suggested as having an influence on evacuation. We therefore subdivide the frequencies of all accidents initiated by seismic events into regional disasters (RDs) (with accelerations greater than 0.4g) and non-regional disasters (with accelerations less than 0.4g.) This classification is carried through to the final probabilities given for each of the failure modes and release paths in Section 5 of this report.

The conditional probabilities associated with the various failure modes and release paths were based on the containment event trees in Reference [2] with a number of modifications that are discussed in Section 2 to this report. The failure modes for external events were taken from the LGS-SARA.

The fission product release fractions were calculated using the MARCH/CORRAL system of codes. The MARCH[13] analysis uses the latest decay heat standard, which results in significantly shorter times to major events in the accident progressions than calculated in References [1] and [2]. The source terms used in the CORRAL[14] analysis are based on the Reactor Safety Study.[9] The use of the RSS source terms is part of the "base-case" approach described in Section 1.1.

1.3 Organization of Report

The binning of the various accident sequences into representative classes is discussed in Section 2. In addition, the determination of the conditional probabilities of the various containment building failure modes is also described. The probabilities of all of the failure modes are determined in Section 2.

In Section 3, the MARCH[13] analysis of the various representative accident sequences is described. This section provides the timing of major events in the accident progression. In addition, the intercompartmental flow is predicted, which in turn determines the movement of fission products (required input to Section 4). Finally, Section 3 provides the basis for the determination of the characteristics of the release from the containment building.

The release of fission products (FPs) from the damaged fuel and the movement of these FPs through the containment building are calculated in Section 4. In addition, Section 4 calculates the fractions of the airborne FPs that are released to the environment when the containment building is predicted to fail.

Finally, in Section 5, the representative source terms for the various failure modes and release paths are generated. Section 5 therefore assembles the information contained in Sections 2 through 4 of this report.

1.4 References to Section 1

1) Philadelphia Electric Company, "Limerick Generating Station, Probabilistic Risk Assessment," March 1981.

2) I. A. Papazoglou, et al., "Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028, February 1983.

3) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," April 1983.

4) L. T. Ritchie, J. D. Johnson, and R. M. Bland, "Calculations of Reactor Accident Consequences Version 2, CRAC 2: Computer Code - User's Guide," NUREG/CR-2326, February 1983.

5) M. A. Azarm, et al., "A Preliminary Review of the Limerick Generating Station Severe Accident Risk Assessment, Volume 1: Core Melt Frequency," NUREG/CR-3493, January 1984.

6) H. Ludewig, J. W. Yang, and W. T. Pratt, "Containment Failure Mode and Fission Product Release Analysis for the Limerick Generating Station: Base Case Assessment," Draft BNL Report dated August 1983.

7) NRC Memorandum from B. Sheron, Branch Chief/RSB to L. Hulman, Branch Chief/AEB, "Set of Release Categories for Limerick DES," dated August 15, 1983.

8) "DES Related to the Operation of the Limerick Generating Station, Units 1 and 2," NUREG-0974, Supplement No. 1, dated December 1983.

9) "Reactor Safety Study: An Assessment of Accident Risk in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/104, 1975.

10) "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," USNRC Report NUREG-0772, June 1981.

11) "Radionuclide Release Under Specific LWR Accident Conditions," Draft BMI-2104 Report, 1983.

12) NRC memorandum from B. Sheron, Branch Chief/RSB to R. Mattson, Director, DSI," Proposed PRA Methodology for Limerick and GESSAR," dated July 22, 1983.

13) R. O. Wooton and H. I. Avci, "MARCH Code Description and User's Manual," NUREG/CR-1711, October 1980.

14) R. J. Burian and P. Cybulskis, "CORRAL 2 User's Manual," BCL report dated January 1977.

2.0 BINNING OF ACCIDENT SEQUENCES

The process of "binning" is a means of reducing a large number of accident sequences into a smaller number of "representative" sequences or classes that can be analyzed to determine potential containment building failure modes. Each of these failure modes will have unique fission product release characteristics. It is intended that the failure modes and fission product release characteristics associated with a particular accident class will be representative of the many individual accident sequences "binned" into the class. In the LGS-PRA,[1] all accident sequences were binned into four generic accident classes (Classes I through IV in Table 2.1). Note that in the LGS-PRA, only accidents initiated by internal events were considered. Containment event trees were then used to determine potential failure modes. The development of the trees and the selection of branch point probabilities depend on a detailed assessment of core meltdown phenomena and the response of the containment building. Seven (reduced from eleven because of similarities) potential failure modes or fission product release paths were identified (refer to Table 2.2) for each of the four classes. The combination of four classes and seven failure modes resulted in a total of twenty-eight distinct fission product release characteristics (source terms). These twenty-eight source-terms were reduced to five for use in the site consequence analysis in the LGS-PRA. The above binning procedure was reviewed extensively at BNL and the details of our review are given in Section 6 of NUREG/CR-3028.[2] This review will not be repeated here, but our basic conclusions will be used later in this section to establish the probabilities of the various failure modes and release paths.

In the LGS-SARA[3] the four generic accident classes (I through IV) were used by PECo whenever possible to represent accident sequences initiated by external events. However, because of the unique characteristics of some of the seismically initiated sequences, two additional accident classes were developed (Classes IS and S in Table 2.1). For Classes IS and S, containment event trees were not necessary (refer to p. 9-4 of Reference 3) because in both cases the containment was assumed to be open from the start of the accident.

At this stage a preliminary review[4] of the LGS-SARA has been performed at BNL which did not include a detailed re-evaluation of the frequencies of the core melt accident sequences. Consequently, it has been necessary to use at face value, significant portions of the LGS-SARA directly as input to the probabilities of the various failure modes and release paths associated with externally initiated accident sequences.

In this section we attempt to calculate the probabilities of the various failure modes. The calculations were started by binning the various accident sequences into representative damage states. This process was done by NRC staff and is reproduced here in Section 2.1. After the damage states were identified, the conditional probabilities of the various failure modes and release paths are determined in Section 2.2. Finally, in Section 2.3, the probabilities of the representative damage states (Section 2.1) and the conditional probabilities of the failure modes (Section 2.2) are combined to give the source term probabilities. These probabilities are used in Section 5 to fully define the source terms for use in the NRC site consequence analysis.

2.1 NRC Staff Classification of Core-Melt Sequences

The classification of core-melt sequences and selection of damage state frequencies was performed by NRC staff and is included in this report only for ease of reference. The process is described in Reference [5] and relies heavily on References [1] through [3]. Therefore, the classifications and frequencies contained in this section and the associated tables should not be regarded as BNL estimates. They were selected by NRC staff with input from BNL reports and other information sources as discussed below.

A complete listing of the higher frequency accident sequences is given in Table 2.3. The accident sequences or damage states are identified by the initiating event (internal, seismic or fire) and by the classes described in Table 2.1. The accident sequences and the determination of their associated probabilities are described in detail in References [1] through [3]. Basically, the "internal" damage states are those evaluated by BNL in NUREG/CR-3028,[2] while the "external" damage states are those described in the LGS-SARA. Note that since Chapters 5, 6, 7 and 8 of the LGS-SARA were not part of the April 1983 version of the documents, contributions to the external events from the initiators in these chapters are not included in Table 2.3.

In response to an NRC memorandum[6] of June 15, 1983, we have subdivided the seismic events damage states into two categories; those that are not classified as regional disasters (RDs) ("g" level less than 0.40) and those classified as RDs ("g" level greater than 0.40). This subdivision is then carried through to the listing of the release categories for incorporation into source term characterization in Section 5.

Based on the damage states listed in Table 2.3, we developed 10 damage state surrogates also listed in Table 2.3. The reduction from 67 damage states to 10 damage states is made possible because many of the original damage states are very similar in terms of the core-melt accident progression and containment failure characteristics. Table 2.4 gives a brief description of each of the surrogate damage states. It should be noted that although LOCAs are considered very low probability events, we have included two in the ten damage states, namely, a small-break, Class I LOCA labeled as I-S and a large-break ATWS-type Class IV LOCA, labeled as IV-A. These are included because they result in sufficiently different release categories to warrant separate consideration. For the "IS" Class accidents, two damage states were chosen (in a similar manner to the LGS-SARA), namely, an ATWS-event, IS-C, and a non-ATWS-event, IS-$\overline{C}$. For the "S" Class accidents, two damage states again were chosen with the distinguishing feature being whether the vessel failure drained the lower plenum water (S-$\overline{H2O}$) or whether it did not (S-H2O).

All ten of the damage states in Table 2.3 have been analyzed using the MARCH (Section 3) and CORRAL (Section 4) computer codes. Each damage state has the potential to fail the containment by a number of failure modes and release paths. In the following section we calculate the conditional probabilities of achieving the various failure modes for each damage state.

Note that the information in Table 2.3 was used as input to the LGS-DES and based on information in References [1] through [3]. However, for input to the LGS-FES, it was decided by NRC staff to revise the probabilities in Table 2.3. The revised numbers are included in Table 2.4 and reflect a change in the frequency for loss-of-offsite-power suggested by BNL.[7] In addition, Table 2.4 also reflects the revised[8] frequency for sequences initiated by fire.

2.2 Containment Failure Modes and Release Paths

In this section we define the conditional probabilities associated with the various containment failure modes and release paths for the ten damage states defined in Table 2.5. For damage states I-S, I-T, II-T, III-T, IV-T and IV-A, the containment event trees developed at BNL[2] were used to calculate the probabilities of the failure modes. However, the BNL event trees were modified for the present analysis. The modifications are described in detail later in this section.

Containment event trees simply relate a given damage state to a number of potential failure modes or release paths. In the present analysis we have defined seven failure modes and include a description of them in Table 2.5. For a given damage state, the containment event trees in Reference [2] can be used to calculate the conditional probabilities of achieving one or more of the failure modes in Table 2.6. The damage states and the conditional probabilities of the failure modes can be summarized in a containment matrix (refer to Table 2.7). A damage state together with one of the failure modes in Table 2.7 defines a unique fission product release path. From an inspection of Table 2.7, it is clear that forty release paths have been defined.

The containment event trees used for damage states I-S, I-T, II-T, III-T, IV-T and IV-A are basically those of Reference [2] with the following modifications. The probability of a steam explosion induced failure of the containment building has been reduced from 10^{-3} to 10^{-4}, which is more consistent with the current trends and beliefs regarding this failure mode. Also, the conditional probability of a failure in the wetwell, which results in suppression pool drainage, has been reduced from the value (CP=0.25) used in Reference [2] to a CP=0.05 as given in Reference [1]. This change is based on a structural assessment of the LGS containment by NRC staff. This assessment found that if a crack occurs in the wetwell, it will tend to propagate upwards, hence the probability of suppression pool drainage should be given a relatively low probability.

2.3 Source-Term Probability

In this section we simply indicate how the probabilities of the damage states were combined with the conditional probabilities of the failure modes to determine the frequency of occurrence of the source terms. Tables 2.3 and 2.4 include the frequencies of each damage state and also indicates the frequencies of accident sequences that result from regional disasters. These probabilities are summarized in Table 2.8 for input of the LGS-DES and in Table 2.9 for input to the LGS-FES.

In Table 2.10 we indicated the assignment of release categories to the conditional probabilities in Table 2.7. Note that the forty potential failure modes have been reduced to twenty-seven release categories because of similarities. The rational for combining the failure modes is discussed further in Sections 4 and 5. The final step is simply to multiply the probabilities of the various damage states in Table 2.8 and Table 2.9 and the corresponding conditional probabilities in Table 2.7 to calculate the frequencies of the twenty-seven release categories in Table 2.10. These frequencies are included in the source term characterization in Section 5.

2.4 References to Section 2

1) Philadelphia Electric Company, "Limerick Generating Station, Probabilistic Risk Assessment," March 1981.

2) I. A. Papazoglou, et al., "Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028, February 1983.

3) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," April 1983.

4) M. A. Azarm, et al., "A Preliminary Review of the Limerick Generating Station Severe Accident Risk Assessment, Volume 1: Core Melt Frequency," Draft BNL report dated August 15, 1983.

5) NRC memorandum from B. Sheron, Branch Chief/RSB to L. Hulman, Branch Chief/AEB, "Set of Release Categories for Limerick DES," dated August 15, 1983.

6) NRC memorandum from L. J. Hulman, Branch Chief/AEB to A. Thadani, Branch Chief/RRAB and B. Sheron, Branch Chief/RSB, "Technical Assistance Request for Severe Accident Analysis for Limerick DES Supplement-RE: SARA," June 15, 1983.

7) I. A. Papazoglou, "Frequency of Loss-of-Offsite-Power in the BNL Review of the Limerick PRA," BNL Memorandum to W. Y. Kato, dated July 19, 1983.

8) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," Supplement No. 2, dated November 1983.

Table 2.1* Generic accident-sequence classes

Generic Accident-Sequence Designator	Physical Basis for Classification	System-Level Contributing Event Sequence
Class I (C1)	Relatively fast core melt; containment intact at core melt and at low pressure	Transients involving loss of inventory make-up, small LOCA events involving loss of in-inventory make-up
Class II (C2)	Relatively slow core melt due to lower decay heat power; containment failed before core melt	Transients or LOCAs involving loss of heat removal, inadvertent SRV opening accidents with inadequate heat-removal capability
Class III (C3)	Relatively fast core melt; containment intact at core melt but a high internal pressure	Transients involving loss of scram function and inability to provide coolant make-up, large LOCAs with insufficient coolant make-up, transient with loss of heat removal and long-term loss of inventory make-up
Class IV (C4)	Relatively fast core melt; containment fails before core melt because of overpressure	Transients that involve loss of scram function and a loss of containment heat removal or all reactivity control but have coolant make-up capability
Class IS	Relatively fast core melt; containment fails before core melt because RHR suction lines are severed	Seismically induced sequences that lead to failure of the inventory make-up systems and a breach of wetwell integrity, with the reactor scrammed
Class S	Relatively fast core melt in an open vessel and failed containment	Reactor-vessel failure with immediate containment failure

*Reproduced from Table 12-4 of Reference 3.

Table 2.2 Failure modes used in the LGS-PRA

Designator	Description
α	Steam explosion in vessel
β, μ'	Steam explosion in containment and H_2 explosion induced containment failure
γ, μ	Overpressure failure - release through drywell and H_2 deflagration sufficient to cause containment overpressure failure
γ'	Overpressure failure - release through wetwell break
γ''	Overpressure failure - wetwell pool drained
$\xi\varepsilon$, $\delta\varepsilon$	Overpressure, large leak and small leak both with SGTS failure
ξ, δ	Overpressure, large leak and small leak both with SGTS operating

Table 2.3 Classification of damage states as used in LGS-DES (core-melt sequences)

Initiating-Event Type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS I		
Internal	1	Table 5.22,23 (Ref. 2)	S_1QUV	7.6-8[a]	I-S
	2	Table 5.26 (Ref. 2)	T_FQUX	3.7-5	(7.6-8)
	3	" "	T_EUV	3.2-5	
	4	" "	T_EUX	8.6-6	
	5	" "	T_TQUX	8.0-6	
	6	" "	T_TUX	4.0-6	
	7	" "	T_T(DC)	2.0-6	
	8	" "	T_FQUV	1.1-6	
	9	" "	T_T(AC)	6.1-7	I-T
	10	" "	T_T(WSW)	6.1-7	(1.0-4)
	11	" "	$T_IC'UX$	5.0-7	
	12	" "	T_IUV	3.6-7	
	13	" "	T_MQUX	3.6-7	
	14	" "	T_F(DC)	3.1-7	
	15	" "	T_F(AC)	9.2-8	
Seismic	16a	Table 12.5 (Ref. 3)	T_SE_SUX[b]	9.0-7	
	16b	Table 12.5 (Ref. 3)	T_SE_SUX(RD)[b]	2.27-6	
Fire	17	Table 12.5 (Ref. 3)	T_FUV	2.3-6	

Table 2.3 Classification of damage states as used in LGS-DES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS II		
Internal	18	Table 5.27 (Ref. 2)	T_FQW	1.3-6	
	19	" "	T_TPW	7.7-7	
	20	" "	T_EW	6.4-7	
	21	" "	$T_T(WSW)$	5.9-7	II-T
	22	" "	T_IW	4.3-7	(4.1-6)
	23	" "	T_FPW	1.2-7	
	24	" "	$T_F(WSW)$	1.1-7	
	25	" "	T_TQW	9.4-8	
Seismic	26a	Table 12.5 (Ref. 3)	T_SE_SW	1.0-8	
	26b	Table 12.5 (Ref. 3)	T_SE_SW (RD)	4.0-8	

Table 2.3 Classification of damage states as used in LGS-DES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS III		
Internal	27	Table 5.28 (Ref. 2)	$T_T{}^1C_MPU$	8.7-7	
	28	" "	$T_FC_M{}^UUR$	5.3-7	
	29	" "	$T_E{}^3C_MW_{12}$	4.3-7	
	30	" "	$T_I{}^4C_MU$	2.9-7	
	31	" "	$T_I{}^4C_MC_{12}$	2.6-7	
	32	" "	$T_F{}^2C_MPU$	2.4-7	
	33	" "	$T_F{}^2C_MW_{12}$	1.6-7	
	34	" "	$T_T{}^1C_MC_2$	1.6-7	
	35	" "	$T_E{}^3C_MUU_R$	1.1-7	III-T
	36	" "	$T_T{}^1C_MPW_2$	6.6-8	(4.1-6)
	37	" "	$T_F{}^2C_MC_2$	4.3-8	
	38	" "	$T_I{}^4C_MPU$	3.2-8	
	39	" "	$T_E{}^3C_MPU$	2.4-8	
	40	" "	$T_E{}^3C_MC_2$	2.1-8	
Seismic	41a	Table 12.5 (Ref. 3)	$T_SRPV\ \overline{R}_B\overline{H}_E$	3.9-8	
	41b	Table 12.5 (Ref. 3)	$T_SRPV\ \overline{R}_B\overline{H}_E(RD)$	3.5-7	
	42a	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2$	4.3-8	
	42b	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2(RD)$	3.9-7	

Table 2.3 Classification of damage states as used in LGS-DES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS IV		
Internal	43	Table 5.21,22, 23 (Ref. 2)	AC	5.0-9	IV-A
	44	" "	$T_T{}^1C_MD$	1.4-7	(5.0-9)
	45	" "	$T_F{}^2C_MUD$	4.0-8	
	46	" "	$T_F{}^2C_MD$	3.3-8	
	47	" "	$T_I{}^4C_MPW_2$	2.0-8	
	48	" "	$T_I{}^1C_ER$	1.5-8	
	49	" "	$T_T{}^1C_MU_H$	1.4-8	
	50	" "	$T_T{}^1C_MPD$	1.3-8	IV-T
	51	" "	$T_T{}^1C_MM$	7.5-9	(4.2-7)
	52	" "	$T_T{}^1C_MR$	7.4-9	
	53	" "	$T_I{}^4C_MD$	3.6-9	
	54	" "	$T_F{}^2C_MPD$	3.6-9	
	55	" "	$T_F{}^2C_MU_H$	3.3-9	
	56	" "	$T_E{}^3C_MPD$	3.0-9	
	57	" "	$T_E{}^3C_MUD$	2.7-9	
	58	" "	$T_E{}^3C_MD$	2.7-9	
	59	" "	$T_F{}^2C_MM$	2.1-9	
	60	" "	$T_T{}^1C_MPU_M$	1.3-9	
Seismic	61a	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2$	1.5-8	
	61b	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2$(RD)	9.5-8	

Table 2.3 Classification of damage states as used in LGS-DES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS IS		
Seismic	62a	Table 12.5 23 (Ref. 3)	$T_S R_B$	1.0-7	IS-$\overline{C}$
	62b	" "	$T_S R_B$(RD)	9.0-7	(1.0-6)
	63a	" "	$T_S R_B C_M$	1.4-8	IS-C
	63b	" "	$T_S R_B C_M$(RD)	1.3-7	(1.4-7)
			CLASS S		
Internal	64	Table 12.5 (Ref. 3)	R	2.7-8	S-H2O
Seismic	65b	" "	$T_S RPVR_B$(RD)		(5.5-8)
	66b	" "	$T_S RPH_E$(RD)		S-$\overline{H2O}$ (3.8-7)
	67b	" "	$T_S RPV\overline{R}\ \overline{H}$(RD)		
Seismic Total				4.1-7	

a $7.6\text{-}8 = 7.6 \times 10^{-8}$

b The seismic damage states have been subdivided into those which also represent "regional disasters" (RD) and those which do not. This separation is for the DES site-consequence analysis.

Table 2.4 Classification of damage states as used in LGS-FES (core-melt sequences)

Initiating-Event Type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS I		
Internal	1	Table 5.22,23 (Ref. 2)	S_1QUV	7.6-8[a]	I-S
	2	Table 5.26 (Ref. 2)	T_FQUX	3.7-5	(7.6-8)
	3	Reference [6]	T_EUV	1.8-5	
	4	" "	T_EUX	4.9-6	
	5	Table 5.26 (Ref. 2)	T_TQUX	8.0-6	
	6	" "	T_IUX	4.0-6	
	7	" "	$T_T(DC)$	2.0-6	
	8	" "	T_FQUV	1.1-6	
	9	" "	$T_T(AC)$	6.1-7	I-T
	10	" "	$T_T(WSW)$	6.1-7	(8.31-5)
	11	" "	$T_IC'UX$	5.0-7	
	12	" "	T_IUV	3.6-7	
	13	" "	T_MQUX	3.6-7	
	14	" "	$T_F(DC)$	3.1-7	
	15	" "	$T_F(AC)$	9.2-8	
Seismic	16a	Table 12.5 (Ref. 3)	$T_SE_SUX(RD)$[b]	9.0-7	
	16b	Table 12.5 (Ref. 3)	$T_SE_SUX(RD)$[b]	2.27-6	
Fire	17	Reference [7]	T_FUV	3.1-6	

Table 2.4 Classification of damage states as used in LGS-FES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS II		
Internal	18	Table 5.27 (Ref. 2)	T_FQW	1.3-6	
	19	" "	T_TPW	7.7-7	
	20	Reference [6]	T_EW	3.7-7	
	21	Table 5.27 (Ref. 2)	$T_T(WSW)$	5.9-7	II-T
	22	" "	T_IW	4.3-7	(3.8-6)
	23	" "	T_FPW	1.2-7	
	24	" "	$T_F(WSW)$	1.1-7	
	25	" "	T_TQW	9.4-8	
Seismic	26a	Table 12.5 (Ref. 3)	T_SE_SW	1.0-8	
	26b	Table 12.5 (Ref. 3)	T_SE_SW (RD)	4.0-8	

Table 2.4 Classification of damage states as used in LGS-FES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS III		
Internal	27	Table 5.28 (Ref. 2)	$T_T{}^1 C_M PU$	8.7-7	
	28	" "	$T_F C_M{}^U UR$	5.3-7	
	29	Reference [6]	$T_E{}^3 C_M W_{12}$	2.5-7	
	30	Table 5.28 (Ref. 2)	$T_I{}^4 C_M U$	2.9-7	
	31	" "	$T_I{}^4 C_M C_{12}$	2.6-7	
	32	" "	$T_F{}^2 C_M PU$	2.4-7	
	33	" "	$T_F{}^2 C_M W_{12}$	1.6-7	
	34	" "	$T_T{}^1 C_M C_2$	1.6-7	
	35	Reference [6]	$T_E{}^3 C_M U U_R$	6.3-8	III-T
	36	Table 5.28 (Ref. 2)	$T_T{}^1 C_M P W_2$	6.6-8	(3.9-6)
	37	" "	$T_F{}^2 C_M C_2$	4.3-8	
	38	" "	$T_I{}^4 C_M PU$	3.2-8	
	39	Reference [6]	$T_E{}^3 C_M PU$	1.4-8	
	40	" "	$T_E{}^3 C_M C_2$	1.2-8	
Seismic	41a	Table 12.5 (Ref. 3)	$T_S RPV\ \overline{R}_B \overline{H}_E$	3.9-8	
	41b	Table 12.5 (Ref. 3)	$T_S RPV\ \overline{R}_B \overline{H}_E (RD)$	3.5-7	
	42a	Table 12.5 (Ref. 3)	$T_S E_S C_M C_2$	4.3-8	
	42b	Table 12.5 (Ref. 3)	$T_S E_S C_M C_2 (RD)$	3.9-7	

Table 2.4 Classification of damage states as used in LGS-FES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS IV		
Internal	43	Tables 5.21,22, 23 (Ref. 2)	AC	5.0-9	IV-A
	44	" "	$T_T^1C_MD$	1.4-7	(5.0-9)
	45	" "	$T_F^2C_MUD$	4.0-8	
	46	" "	$T_F^2C_MD$	3.3-8	
	47	" "	$T_I^4C_MPW_2$	2.0-8	
	48	" "	$T_T^1C_ER$	1.5-8	
	49	" "	$T_T^1C_MU_H$	1.4-8	
	50	" "	$T_T^1C_MPD$	1.3-8	IV-T
	51	" "	$T_T^1C_MM$	7.5-9	(4.2-7)
	52	" "	$T_T^1C_MR$	7.4-9	
	53	" "	$T_I^4C_MD$	3.6-9	
	54	" "	$T_F^2C_MPD$	3.6-9	
	55	" "	$T_F^2C_MU_H$	3.3-9	
	56	Reference [6]	$T_E^3C_MPD$	1.7-9	
	57	" "	$T_E^3C_MUD$	1.5-9	
	58	" "	$T_E^3C_MD$	1.5-9	
	59	Tables 5.21,22, 23 (Ref. 2)	$T_F^2C_MM$	2.1-9	
	60	" "	$T_T^1C_MPU_M$	1.3-9	
Seismic	61a	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2$	1.5-8	
	61b	Table 12.5 (Ref. 3)	$T_SE_SC_MC_2$(RD)	9.5-8	

Table 2.4 Classification of damage states as used in LGS-FES (core-melt sequences) (Cont.)

Initiating-event type	Number of Damage State	Cross Reference	Damage States	Annual Frequency (point estimate)	Damage State Surrogates
			CLASS IS		
Seismic	62a	Table 12.5 23 (Ref. 3)	$T_S R_B$	1.0-7	IS-$\overline{C}$
	62b	" "	$T_S R_B$(RD)	9.0-7	(1.0-6)
	63a	" "	$T_S R_B C_M$	1.4-8	IS-C
	63b	" "	$T_S R_B C_M$(RD)	1.3-7	(1.4-7)
			CLASS S		
Internal	64	Table 12.5 (Ref. 3)	R	2.7-8	S-H2O
Seismic	65b	" "	$T_S RPVR_B$(RD)		(5.5-8)
	66b	" "	$T_S RPH_E$(RD)		S-$\overline{H2O}$ (3.8-7)
	67b	" "	$T_S RPV\overline{R}\ \overline{H}$(RD)		
Seismic Total				4.1-7	

a $7.6\text{-}8 = 7.6 \times 10^{-8}$

b The seismic damage states have been subdivided into those which also represent "regional disasters" (RD) and those which do not. This separation is for the DES site-consequence analysis.

Table 2.5 Description of damage states

Designator	Description
I-S	These are LOCA initiated sequences (medium and small breaks only) involving loss of inventory make-up. They result in a relatively fast core melt and the containment is intact at the time of core melt.
I-T	These are sequences initiated by transients again involving loss of inventory make-up. Core melt is relatively fast and the containment is intact at the time of core melt.
II-T	These are transient or LOCA initiated sequences involving loss of containment heat removal or inadvertent SRV opening accidents with inadequate heat removal capability. Core melt is relatively slow due to lower decay power level and the containment has failed prior to core melt.
III-T	Transients involving loss of scram function and inability to provide coolant make-up, large LOCAs with insufficient coolant make-up, transients with loss of heat removal and long-term loss of inventory make-up. Core melt is relatively fast and the containment is intact at core melt.
IV-T	Transients that involve loss of scram function and a loss of containment heat removal or all reactivity control but have coolant make-up capability. Core melt is relatively fast and the containment fails prior to core melt because of overpressure.
IV-A	As above but initiated by large LOCAs.
IS-$\overline{C}$	Seismically induced sequences that lead to failure of the inventory/make-up systems and a breach of wetwell integrity, with the reactor scrammed. Core melt is fast and the containment fails prior to core melt because the RHR suction lines are severed.
IS-C	As above but coupled with a loss of the scram function.
S-H2O	Seismically induced reactor-vessel failure (plus random reactor-vessel failure) coupled with immediate containment failure. Core melt is fast and the vessel and containment are both failed at the time of core melt. This sequence assumes the vessel break is high, which allows water to be retained in the bottom of the vessel prior to core slump.
S-$\overline{H2O}$	As above but with a vessel failure location that results in complete draining of the water from the vessel.

Table 2.6 Containment failure mode and release path notation

Designator	Description
DW	Containment Failure via overpressurization. Failure location in the drywell.
WW	Containment Failure via overpressurization. Failure location in the wetwell above the suppression pool.
$\overline{WW}$	Containment Failure via overpressurization. Failure location in the wetwell below the suppression pool resulting in loss of suppression pool water.
SE	Failure via in-vessel steam explosion generated missiles.
HB	Failure via H_2 burning during the periods when the containment atmosphere is deinerted. This failure mode also includes H_2 detonation and ex-vessel steam explosion failure modes, which are of very low frequency.
LGT	Containment leakage rates sufficiently low to allow the standby gas treatment system (SGTS) to operate effectively.
$L\overline{GT}$	Containment leakage rates so high that the SGTS is ineffective.

Table 2.7 Assignment of conditional probabilities

Damage States	Containment Failure Modes and Release Paths							
	DW	WW	$\overline{WW}$	SE	HB	LGT	$L\overline{GT}$	No Core Melt
I-S	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
I-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
II-T	0.250	0.225	0.025	0.0001	0	0	0	0.5
III-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
IV-A	0.500	0.45	0.05	0.0001	0	0	0	0
IV-T	0.500	0.45	0.05	0.0001	0	0	0	0
IS-C	1*	0	0	0.0001	0	0	0	0
IS-$\overline{C}$	1*	0	0	0.0001	0	0	0	0
S-H20	0	0	1**	0.0001	0	0	0	0
S-$\overline{H20}$	0	0	1**	0	0	0	0	0

*In the LGS-SARA, this failure mode was considered similar to a drywell (DW) failure mode, however, this should not be interpreted as a failure location in the drywell. Class IS sequences result in failure of the RHR suction lines, which partially drains the suppression pool exposing the downcomers but leaving the quenchers submerged. Thus, the melt release will be scrubbed by the pool (similar to WW failure mode) and the vaporization release will not be scrubbed by the pool (similar to DW failure mode).

**Again, assigning the $\overline{WW}$ failure mode to Class S sequences relates to the fission product release path (and lack of suppression pool scrubbing) rather than to the failure location.

Table 2.8 Damage state probabilities as used in LGS-DES

Damage State	Total Probability	Probability Regional Disasters	Probability Non-Regional Disasters
I-S	7.6(-8)*	-	7.6(-8)
I-T	1.0(-4)	2.27(-6)	9.8(-5)
II-T	4.1(-6)	4.0(-8)	4.06(-6)
III-T	4.1(-6)	7.4(-7)	3.36(-6)
IV-A	5.0(-9)	-	5.0(-9)
IV-T	4.2(-7)	9.5(-8)	3.25(-7)
IS-C	1.44(-7)	1.3(-7)	1.4(-8)
IS-$\overline{C}$	1.0(-6)	9.0(-7)	1.0(-7)
S-H20	5.45(-8)	4.1(-8)	1.35(-8)
S-$\overline{H20}$	3.83(-7)	3.79(-7)	1.35(-8)

*7.6(-8) = 7.6×10^{-8}

Table 2.9 Damage state probabilities as used in LGS-FES

Damage State	Total Probability	Probability Regional Disasters	Probability Non-Regional Disasters
I-S	7.6(-8)*	-	7.6(-8)
I-T	8.31(-5)	2.27(-6)	8.1(-5)
II-T	3.8(-6)	4.0(-8)	3.8(-6)
III-T	3.9(-6)	7.4(-7)	3.2(-6)
IV-A	5.0(-9)	-	5.0(-9)
IV-T	4.2(-7)	9.5(-8)	3.25(-7)
IS-C	1.44(-7)	1.3(-7)	1.4(-8)
IS-$\overline{C}$	1.0(-6)	9.0(-7)	1.0(-7)
S-H20	5.45(-8)	4.1(-8)	1.35(-8)
S-$\overline{H20}$	3.83(-7)	3.79(-7)	1.35(-8)

*7.6(-8) = 7.6×10^{-8}

Table 2.10 Assignment of release categories

Damage States	Containment Failure Modes and Release Paths							
	DW	WW	$\overline{WW}$	SE	HB	LGT	L$\overline{GT}$	No Core Melt
I-S	I-S/DW	I-S/DW	I-S/DW	I-T/SE	I-T/HB	I-T/LGT	I-T/L$\overline{GT}$	n/a
I-T	I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE	I-T/HB	I-T/LGT	I-T/L$\overline{GT}$	n/a
II-T	II-T/WW	II-T/WW	II-T/WW	II-T/SE	n/a	n/a	n/a	No Core Melt
III-T	III-T/WW	III-T/WW	III-T/WW	III-T/SE	III-T/HB	III-T/LGT	III-T/L$\overline{GT}$	n/a
IV-A	IV-A/DW	IV-A/DW	IV-A/DW	IV-T/SE	n/a	n/a	n/a	n/a
IV-T	IV-T/DW	IV-T/WW	IV-T/$\overline{WW}$	IV-T/SE	n/a	n/a	n/a	n/a
IS-C	IS-C/DW	n/a	n/a	IS-C/SE	n/a	n/a	n/a	n/a
IS-$\overline{C}$	IS-$\overline{C}$/DW	n/a	n/a	IS-$\overline{C}$/SE	n/a	n/a	n/a	n/a
S-H20	n/a	n/a	S-H20/$\overline{WW}$	S-H20/SE	n/a	n/a	n/a	n/a
S-$\overline{H20}$	n/a	n/a	S-$\overline{H20}$/$\overline{WW}$	n/a	n/a	n/a	n/a	n/a

3.0 CORE MELTDOWN AND CONTAINMENT RESPONSE

In this section, MARCH[1] analyses of core meltdown and containment response for the various representative accident sequences are presented. The MARCH computer code has been modified to include a new decay heat model based on the ANS-5.1-1979[2] standard. The new MARCH computer code model (programmed by C. Shaffer of Sandia National Laboratories) includes the decay of U239 and NP239 and the effect of neutron capture in fission products which was not included in the original MARCH model. The new model produces an integrated decay heat over the first hour after shutdown about 20% greater than the original MARCH model, which was used in the previous BNL review[3] of the LGS-PRA.[4] The main effect of the new decay heat model is the change in timing of major events. The time to core meltdown, reactor pressure vessel (RPV) failure and containment failure predicted using the new decay heat model are significantly earlier than that in NUREG/CR-3028. The highlights of the basic assumption of the MARCH analysis are summarized in Table 3.1. This table is similar to Table 7.12 in NUREG/CR-3028. The major difference relates to the assumed decontamination factor (input parameter DCF in MARCH) for a saturated pool. In NUREG/CR-3028 a DCF=10 was assumed for a saturated pool, which is consistent with the LGS-PRA. However, in the present study, a DCF=1 is assumed, which is consistent with the CORRAL analysis in Section 4 and our "base case technology" approach (refer to Section 1.2).

3.1 Class I Sequences

The Limerick PRA describes this Class as follows:

> "The Class I (C1) events can be characterized as transients involving loss-of-coolant make-up to the reactor core. For the Limerick analysis, these events are found to have the highest calculated frequency of occurrence. They involve successful control rod insertion; however, there is postulated to be a loss of both high pressure and low pressure injection. The physics model used in the consequence calculation represents the sequence designated TQUV."

The TQUV sequence is analyzed for three containment failure modes, namely, structural failure in the drywell, in the airspace of the wetwell, and below the pool waterline in the wetwell (which drains the suppression pool). Due to the loss-of-coolant make-up to the reactor core, the MARCH code predicts the start of core melting at 90 minutes and slumping into the lower vessel head at about 145 minutes. The molten core is discharged onto the diaphragm floor at 174 minutes. Containment failure is assumed to occur as the core debris penetrates 70 cm of the diaphragm floor. Table 3.2 shows a comparison of the present BNL analysis with the analyses of the Class I (TQUV) sequence in the LGS-PRA and in NUREG/CR-3028 of the Class I (TQUV) sequence. The earlier timing of the major events predicted by this study is caused by using the 1979 decay power standard. During the transient, the suppression pool was predicted to remain subcooled. Hence, a DCF=100 is used in the MARCH analysis. It is noted that the vessel head failure occurs prior to the containment failure. Any rapid debris/water interaction or steam explosion in the pedestal region takes place in an inerted atmosphere. It was assumed that the inerted atmosphere prevents the oxidation release associated with steam explosions (refer to Section 4).

To simulate a failure in the wetwell in which the suppression pool is drained into the reactor enclosure, the MARCH input parameter NT is specified as -7. Using this option, the MARCH code assumes[5] complete failure of the suppression pool. The intercompartment transfers from the drywell to the wetwell and direct blowdown from the primary system through SRV's are assumed to bypass the suppression pool.

In addition to the transient events discussed above, the LGS-PRA also considered accidents initiated by loss-of-coolant accidents (LOCAs). For Class I sequences, an S_1QUV (medium LOCA) accident was identified as one of the more probable LOCA-initiated sequences. A medium LOCA event is defined as a break of between 0.044 and 0.1 ft^2 in a liquid line, and between 0.016 and 0.08 ft^2 in a steam line. The S_1QUV sequence is defined as a failure of the condensate and feedwater system, and the failure of both high pressure and low pressure injection systems. A MARCH analysis of the S_1QUV has been performed assuming a 0.08 ft^2 break in a steam line. The results are shown in Table 3.3. The timing of the major events are similar to that of the TQUV sequence, which is also included in Table 3.3.

3.2 Class II Sequences

This class is described in Chapter 3 of the LGS-PRA as follows:

> "For Class II (C2), the sequence modeled is a transient with long-term loss of heat removal (TW). For Limerick, this sequence involves the failure of the power conversion system and of the RHR system. Also included in this class are other sequences, such as LOCAs accompanied by a failure of the containment heat removal systems, and inadvertently open relief valves with failure to remove heat from containment. The key feature in this class is that the containment is assumed to fail prior to core melt, but after a lengthy period of time into the accident. Postulated core melt begins with a relatively low decay heat source, leading to a slower core melt than anticipated for Classes I, III, or IV, but with a failed containment."

In our modeling of TW, the high pressure ECC switches its suction from the condensate storage tank (CST) to the suppression pool at ~20 minutes because of a high suppression pool level and fails at 372 min because of pool heatup (temperature > 200 F). The low pressure pumps start at 372 min (TW=TWLP). ECC pumps are throttled back when the total water mass in the pressure vessel exceeds 600,000 lb, hence keeping the water level steady and considerably above the core. Eventually, the containment pressure exceeds 155 psia as steam passed through the SRVs heats up the pool to saturation at about 380 min and then passes the decay heat mostly to the containment atmosphere. Injection is assumed to fail when containment fails. The pool DF is assumed to be 1, because the core melt begins after the pool has been heated to saturation.

Table 3.4 compares the results of the present BNL calculations and with the LGS-PRA and NUREG/CR-3028 results. The timing of the major events predicted by this study are much faster than the predictions of both the LGS-PRA and NUREG/CR-3028. This is because of the pronounced effect of the 1979 decay heat correlation. The increased decay heat causes the reactor pressure to fail at 26.8 hours compared with 38.7 hours in NUREG/CR-3028.

3.3 Class III Sequences

In this transient initiated accident sequence, the control rods fail to insert followed by poison injection failure. The recirculation pumps trip and the feedwater flow is stopped, which rapidly brings the power level down to an assumed 30%. The high pressure injection systems are modeled to come on with flows of 600 gpm (RCIC) and 5600 gpm (HPCI). At 4.5 minutes, the high pressure pump suction is automatically switched from the CST to the suppression pool because of high water level in the suppression pool (ECCRC=0.850).

These high pressure systems will subsequently fail either because of high suppression pool pressure (Limerick PRA assumption) or high (200°F) suppression pool temperatures (BNL assumption). The high pool temperature causes the ECC turbine lubricating oil, which is cooled by the suppression pool water, to break down causing the turbine to seize. The present analysis has the ECC pumps failing at 14 minutes because of this loss of lube oil cooling. Other important MARCH input data for the ATWS-III sequence are:

- Fourteen safety relief valves operate compared to four for Class I.
- MARCH modeling simulates 30% power for the fraction of the core that is covered and ANS-1979 standard decay power for the uncovered part.
- Ex-vessel core debris/water interactions are included but the amount of water on the diaphragm floor is so small that essentially no time delay is observed.

A comparison of the present calculations for the above sequence with the LGS-PRA and NUREG/CR-3028 results is given in Table 3.5.

3.4 Class IV Sequences

The MARCH modeling for the ATWS Class IV sequence differs from the ATWS Class III modeling only in the following respects:

a) The high pressure injection is allowed to stay on even after the suppression pool temperature exceeds 200°F [the ECC turns off when the containment fails - a MARCH parameter option (ICBRK=0)].

b) The choice of containment break area is 5 ft^2 in order to prevent the containment from appreciably overshooting the 155 psia failure pressure. The Limerick PRA used a 3.14 ft^2 hole size.

The ATWS-IV sequence was analyzed for the three containment failure locations considered in Section 3.1. The timing of the major events as predicted in the present analysis, are compared with the LGS-PRA and NUREG/CR-3028 results in Table 3.6.

In addition to a transient initiated event, a large LOCA initiated event (AC) was also analyzed. The sequence AC was identified in the LGS-PRA and is defined as a pipe break in either the recirculation line (water break) or main steam line (steam break), coupled with the failure of the control rods to insert. For our analysis of this sequence, blowdown data for a main steam line break (given in Table 6-2-11 of the LGS-FSAR) were used as input data to the MARCH code. The FSAR blowdown data indicates that the steam and liquid flow rates approach zero in approximately 60 seconds and do not change significantly during the remainder of the sequence. The containment is assumed to fail in the drywell. The results are shown in Table 3.7 and were used as input to the fission product release calculations (Section 4.4.2) for the LGS-DES. However, we noted in Section 1.0 that BNL was requested to reanalyze the Class IV sequences for input to the LGS-FES. The reanalysis is described in the following section.

3.5 Class IV Reanalysis

In the MARCH code, the containment response after the failure of reactor vessel is analyzed by using the INTER[1] code as a subroutine to model corium/concrete interactions. The INTER code was developed in 1977 to model small-scale experiments in which melts with a generally high metallic content were poured into a concrete crucible. In these experiments, the principal attack on the concrete was from the metallic layer that did not wet the concrete and resulted in a purely thermal attack. This is not necessarily the case with oxide melts that do wet the concrete. The applicability of the INTER model to melts having a high oxide content was questioned by the developer of the code, W. B. Murfin. Murfin also stressed that the model represented only a first stage in the modeling project, and he cautioned against applying the model to prototypical containment building melt-through analysis. An improved core/concrete interaction model, CORCON,[6] has been issued by Sandia National Laboratories. The CORCON model roves upon the preliminary INTER model because it is intended to provide quantitative estimates of full-scale reactor fuel-melt accidents. While it is outside the scope of this report to replace the INTER model in MARCH with CORCON, it is possible to run MARCH and CORCON concurrently.

The MARCH/CORCON technique was applied to the four ATWS-IV sequences. The initial conditions for corium/concrete interactions obtained from the MARCH calculations were input to CORCON. The outputs from CORCON, involving the flow rates and temperatures of the steam, hydrogen, carbon dioxide, and carbon monoxide, were fitted as polynominal equations by the least-square method. These polynominal equations were incorporated into the MARCH code which bypassed the INTER model. The CORCON code predicts significantly slower concrete erosion velocities and gas generation rates than the INTER code. Hence, the predicted containment pressure, temperature, and leakage rates are lower than the predictions given by the MARCH/INTER analyses. However, the timing of the major events for the ATWS-IV sequences is not influenced by using the MARCH/CORCON approach. In addition, the following two improvements were added in the revised calculation:

a) Containment break area was reduced from 5 ft^2 to 3 ft^2 to be consistent with the LGS-PRA analysis.

b) Containment heat sinks were increased from 8 to 15 to provide a more realistic representation of all structures in the LGS containment.

The impact of these improvements in the fission products transport and release will be discussed in Section 4.4.3.

3.6 Class IS Sequences

The LGS-SARA[7] describes this class as follows:

"Class IS includes earthquake-initiated transients with a loss of the ability to maintain core-coolant inventory. The reactor-coolant system and the containment structure are otherwise intact. The reactor enclosure is severely damaged by the earthquake. The earthquake also causes the 24-in. RHR suction line from the suppression pool to shear outside the containment. This allows the suppression pool to drain to the level of the RHR suction line. The depth of water above the quenchers is reduced from 18 ft. 11 in. (minimum water level) to 7 ft. The sheared line also provides a direct flow path from the containment to the environment."

The typical sequence for this class was defined in the LGS-SARA as T_SR_B. For this sequence, coolant inventory make-up is lost due to loss of all ac and dc power and the RHR heat-exchanger lines are assumed to be severed. Another sequence $T_SR_BC_M$ is similar to the T_SR_B sequences, but with the reactor also failing to scram. For this class, the suppression pool is available for scrubbing the melt release since the quenchers are below the level of the RHR suction line. The pool DF is 100 as the pool temperature remains subcooled during the transient. However, as the suppression pool has drained to below the level of the downcomers, the vaporization release is not subjected to pool scrubbing.

Both the T_SR_B and $T_SR_BC_M$ sequences have been analyzed and the results are summarized in Table 3.8. No comparisons with LGS results and the previous BNL results are included in Table 3.5. The results of a core meltdown analysis for this class are not presented in the LGS-SARA report and NUREG/CR-3028 did not, of course, consider any externally initiated accidents. The $T_SR_BC_M$ sequence is similar to the T_SR_B sequence but with the reactor failing to scram, the major events of the $T_SR_BC_M$ sequence occur sooner than the T_SR_B sequence.

3.7 Class S Sequences

The LGS-SARA describes this class as follows:

"Class S has two components. The first is an earthquake-induced failure of the reactor-vessel lateral supports, leading to a failure of the main steam lines and a simultaneous containment failure, either directly, through a failure of the RHR suction lines, as described for Class IS, or indirectly, through an overpressurization or a mechanical impact resulting from the vessel losing its lateral restraint. The second is a random reactor-vessel failure accompanied by a containment failure."

There are two source terms appropriate for this class of accident. The two source terms are associated with two cases, one in which vessel failure leads to a loss of all water from the vessel and one in which some water remains in the bottom of the vessel. The source terms for the cases with and without water in the lower plenum are labeled S-H2O and S-$\overline{\text{H2O}}$, respectively (refer to Section 2). A major contributor to this class is identified as the T_SRPVRB sequence.

To simulate the T_SRPVRB sequence, the failure of the main steam lines are considered equivalent to a large LOCA, which occurs simultaneously with the containment failure at the initiation of the accident. The blowdown data for a main steam line break given in Table 6.2-11 of the LGS-FSAR were used as the input data for the MARCH code. The cases associated with the two source terms, S-H2O and S-$\overline{\text{H2O}}$, are modeled in the MARCH code by varying the input parameter WATBH, which is defined as the mass of water that can be stored in the bottom vessel head. The parameter WATBH is taken as 182710 and 100 for the two source terms, S-H2O and S-$\overline{\text{H2O}}$, respectively. (A zero-mass in the bottom head will lead to an overflow condition for the MARCH numerical computation procedure; hence, it is necessary to specify 100 lb of water for the S-$\overline{\text{H2O}}$ case). The results of T_SRPVRB sequence are shown in Table 3.9. Again, we are unable to compare the results in Table 3.9 with the LGS-SARA.

3.8 Summary

MARCH analyses for the various representative accident sequences have been performed. The analyses cover both internal and external initiated accidents and three potential containment failure modes. Using the 1979 decay heat standard, the MARCH predicted timing of major events in the accident progressions are significantly earlier than those reported in NUREG/CR-3028. The MARCH results are summarized in Table 3.10.

3.9 References to Section 3

1) R. O. Wooton and H. I. Avci, "MARCH Code Description and User's Manual," Battelle Columbus Laboratories/USNRC Report NUREG/CR-1711, October 1980.

2) ANSI/ANS-5.1, "Decay Heat Power in Light Water Reactors," August 1979.

3) I. A. Papazoglou, et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory/USNRC Report NUREG/CR-3028, February 1983.

4) Philadelphia Electric Company, "Limerick Generating Station, Probabilistic Risk Assessment," March 1981.

5) S. R. Greene, "Undocumented MARCH BWR Containment Modeling Feature," ORNL memorandum dated January 21, 1983.

6) J. F. Muir et al., "CORCON-MOD1: An Improved Model for Molten-Core/Concrete Interactions," NUREG/CR-2142, July 1981.

7) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," April 1983.

Table 3.1 Highlights of MARCH analysis

1.	Only 2 compartments modeled (wetwell and drywell).
2.	Failure of coolant injection due to overheating of lube oil when the suppression pool temperature is greater than 200 F.
3.	No H_2 burning or detonation.
4.	8 heat sinks used* in MARCH instead of the 17 used in INCOR.
5.	Heat transfer coefficient between steel and concrete = 2 Btu/hr/ft^2.
6.	Pool decontamination factor, DCF = 100 for subcooled water and 1 for saturated water or $\overline{WW}$ failure mode.
7.	Wetwell compartment volume is airspace only [VC(2)=155,000 ft].
8.	Containment failure occurs when penetration of diaphragm floor $\geq$ 70 cm or containment pressure $\geq$ 155 psia.
9.	Containment leakage taken as 1/2% volume/day.
10.	Equivalent clad thickness includes zirconium from fuel channels.
11.	Core slumps when 80% of core is melted.
12.	HOTDROP subroutine made inactive by using MARCH options.
13.	Core debris assumed retained on diaphragm floor inside the pedestal wall.

*8 heat sinks were utilized in the MARCH analysis for the LGS-DES. However, the heat sinks were increased to 15 for the four additional Class IV sequences reanalyzed for the LGS-FES (refer to Section 3.5).

Table 3.2 Comparison of BNL and Limerick PRA analysis of the Class I sequences (TQUV)

Key Events	Analysis in Limerick PRA	BNL Analysis	
		NUREG/CR-3028	This Work
Start of core melt (hours)	1.3	1.65	1.50
Core slump (hours)	2.5	3.08	2.42
Vessel head failure (hours)	4.3	3.71	2.90
Start of core/concrete interactions (hours)	4.3	3.71	2.90
Time (hours) core debris penetrates 70 cm of diaphragm floor causing collapse of floor and containment failure	6.5	6.12	5.17
Pressure at containment failure (psia)	88	113	118

Table 3.3 BNL analysis of S_1QUV sequence

Key Events (hours)	S_1QUV	TQUV
Start of core melt	1.35	1.50
Core slump	2.24	2.42
Vessel head failure	2.83	2.90
Containment failure*	6.0	5.17

*Containment failure caused by 70-cm penetration of the floor, the containment pressure at floor failure is 122 psia for the S_1QUV sequence and 118 for the TQUV sequence.

Table 3.4 Comparison of BNL and Limerick PRA analysis of the Class II sequences (TWLP)

Key Events	Analysis in Limerick PRA	BNL Analysis Ref. [3]*	BNL Analysis This Work
Containment failure (hr)	30	29.2	19.6
Core melt begins (hr)	36.6	36.0	24.9
Core melt ends (hr)	39.0	38.6	26.8
Vessel head fails (hr)	40.8	38.7	26.8
Z = 70-cm penetration (hr)	43.3	47	33.2

*Containment failure mode is a WW and the break area is 0.208 ft^2.

Table 3.5 Comparison of BNL and Limerick PRA analysis for Class III, ATWS sequence

Key Events	Analysis in Limerick PRA	BNL Analysis DF=10 Ref[3]	BNL Analysis DF=1 Ref[3]	BNL Analysis DF=1 This Work
Core melt begins (hr)	0.85	0.78	0.76	0.5
Core melt ends (hr)	2.5	2.30	2.22	1.80
Vessel head failure (hr)	4.3	2.55	2.47	2.05
INTER begins (hr)	4.3	2.55	2.47	2.05
Containment failure (hr)	6.5	4.45*	3.83**	2.67*

*Containment fails because of overpressure (155 psia) before 70-cm of concrete penetration is reached.

**Time that the core debris penetrates 70-cm of diaphragm floor causing floor collapse and containment failure.

Table 3.6 Comparison of BNL and Limerick PRA analysis for Class IV

Key Events	Analysis in Limerick PRA	BNL Analysis DF=10 Ref[3]	DF=1 Ref[3]	DF=1 This Work
Containment fails (hr)	0.67	0.67	0.67	0.67
Core melt begins (hr)	1.2	1.25	1.25	1.16
Core melt ends (hr)	2.2	2.7	2.7	2.24
Vessel head fails (hr)	4.0	2.97	2.95	2.47
Time for 70-cm penetration of floor (hr)*	6.5	6.97	7.03	6.16

*Limerick assumed the molten core to spread over the entire diaphragm floor. BNL assumed the core materials to be confined to the pedestal region.

Table 3.7 BNL analysis of AC sequence of Class IV accident

Key Events (hours)	AC	ATWS
Containment failure	0.65	0.67
Start of core melt	1.17	1.16
Core slump	1.58	2.24
Vessel head failure	2.20	2.47
70-cm penetration of floor	5.43	6.16

Table 3.8 BNL analyses for Class IS

Key Events (hours)	T_SR_B	$T_SR_BC_M$
Containment fails	0	0
Core melt begins	1.47	0.37
Core melt ends	2.32	1.28
Vessel head fails	2.37	1.53
70-cm penetration of floor	6.46	5.02

Table 3.9 BNL analyses of Class S (T_SRPVRB)

Key Events (hours)	S-H2O	S-$\overline{\text{H2O}}$
Containment fails	0	0
Core melt begins	2.67	2.83
Core melt ends	3.65	3.85
Vessel head fails	5.23	4.38
70-cm penetration of floor	8.82	7.22

Table 3.10 Summary of MARCH results

Sequence	Occurrence of Major Events (hr)					Pool Temperature*	Pool DF	Containment Break Area (ft^2)
	Start Melt	Core Slump	Vessel Failure	Containment Failure	70-cm Floor Penetration			
TQUV	1.5	2.42	2.90	5.17	5.17	Subcool	100	5
S_1QUV	1.35	2.24	2.83	6.0	6.0	Subcool	100	5
TWLP	24.9	26.8	26.8	19.6	33.2	Saturation	1	0.208
ATWS	0.5	1.80	2.05	2.67	5.13	Saturation	1	2
ATWS	1.16	2.24	2.47	0.67	6.16	Saturation	1	5**
AC	1.17	1.58	2.20	0.65	5.43	Saturation	1	5
TSRB	1.47	2.32	2.37	0	6.30	Subcool	100	5
TSRBCM	0.37	1.28	1.53	0	4.81	Subcool	100	5
TSRPVRB (S-H2O)	2.67	3.65	5.23	0	8.82	Subcool	1	5
TSRPVRB (S-$\overline{H2O}$)	2.83	3.85	4.38	0	7.22	Subcool	1	5

*Pool temperature during core meltdown.
**A containment break area of 3 ft^2 was used in the Class IV reanalyses (refer to Section 3.5).

4.0 FISSION PRODUCT TRANSPORT AND RELEASE

Due to current activities in the field of fission product chemistry and transport, a two-pronged approach is being pursued in determining the fission product release for core meltdown accidents in the LGS. The two approaches are described in Reference 1, which also gives the justification for the approaches. The two approaches are briefly described below:

1) Base case technology. This approach is based on the Reactor Safety Study[2] (RSS) methods regarding the fission product source terms, pool scrubbing and fission product transport.

2) Advanced technology. This approach will be based on the current methods[3] and data generated by the Accident Source Term Project Office (ASTPO) of NRC/RES.

Briefly, these two approaches will affect the following four areas related to the determination of fission product release:

1) Fission product released from core material. In the base case technology, these quantities are based on the four release periods used in the RSS (gap, melt, oxidation, and vaporization). Furthermore, the timing of the releases will follow the same prescription outlined in the RSS. In the advanced technology case, the release fraction in the reactor pressure vessel (RPV) will be based on the core heatup history and the latest data on fission product release from heated core material. In the ex-RPV release phase, the prediction of fission product release will be determined by models based on the latest core/concrete interaction data.

2) Release of fission products from the RPV to containment building. In the base case technology, no attenuation of the fission products is allowed in the primary system. Thus, all the fission products released during the Gap and Melt release phases enter the containment building. In the advanced technology case, an attempt will be made to determine the fraction of the fission products which after release from the fuel, either plate out or chemically afix themselves to structures in the primary system. This determination will also include that fraction of the retained fission products which are re-emitted, and the timing of the re-emission.

3) Fission product attenuation in the suppression pool. In the base case technology, the suppression pool attenuation will be determined by RSS suggested methods, i.e., a decontamination factor (DF) of 100 is used for the subcooled pools and a DF of 1 is used for the saturated pools. Noble gases and organic iodine are not subject to pool scrubbing. In the advanced technology case, the DF will be determined by a model which will account for parameters such as aerosol particle diameter and density, bubble size and velocity, pool temperature and carrier gas.

4) Fission product transport and atmospheric release. In the base case technology, the fission product transport within the containment building volumes is predicted using the CORRAL-II[4] code. This code is used in conjunction with the fission product release model, pool scrubbing model and the MARCH[5] code as described in Section 3. In the advanced technology case, an upgraded code for fission product transport within the containment will be used. This code will interface with fission product sources from the in-vessel melt release phase and the ex-vessel core/concrete interactions. In addition, mechanistically determined pool DF's will be used. All these quantities will be consistent with the latest methods concerning these phenomena.

In the analysis to be presented in this report, only the base case technology will be used. Thus, the CORRAL-II code with its four distinct core release mechanisms (Gap, Melt, Oxidation, and Vaporization) together with the RSS source model and pool DF model, will be used to determine the fission product transport within the containment. The four release mechanisms are shown schematically in Figure 4.1 (note that the oxidation release was assumed to result from a steam explosion in the RSS.) The gap release is modeled as a single event and is assumed to occur at accident initiation. The melt release is divided into 10 equally sized releases evenly spaced between the time of core melt to the time of core slump. The timing of core melt and slumping were taken directly from the MARCH analysis. The oxidation release is modeled as a single event and chosen to occur at RPV head failure to model the oxidation of that fraction of the core debris assumed to interact with water on the diaphragm floor or to fall into the suppression pool. The vaporization release is divided into 20 parts, 10 releases of exponentially decreasing magnitude in the first 1/2 hour, followed by 10 more releases during the next 1-1/2 hours, also of exponentially decreasing magnitude. The vaporization release is assumed to start after vessel failure when core/concrete interactions begin. The core release fractions for input to CORRAL were obtained from the RSS. Table 4.1 is reproduced from the RSS and indicates the fraction of fission products released corresponding to the release mechanisms noted above. The fractional release of fission products indicated in Table 4.1 would be input to CORRAL using the schematic indicated in Figure 4.1.

For our purposes we use the oxidation release to model the oxidation release when a fraction of the core is assumed to drop into the suppression pool, and the containment building is assumed to be failed at the time of RPV failure. The failed containment building ensures the presence of oxygen which is necessary for the oxidation release to occur. The oxidation release affects only the Kr, Xe, I_2, Te, and Ru releases as assumed in WASH-1400.

In the LGS-PRA,[6] an oxidation release was allowed for at the time of RPV failure for all sequences, and additionally 15% of the suppression pool water was assumed to flash at the time of containment failure. The flash release affects all the isotopes in the suppression pool equally for Classes I and III. Since the fission products in the suppression pool will primarily be from the melt and gap release, the flash release will affect elemental iodine, cesium and barium, more than the remaining fission product groups. Finally, in the LGS-PRA, the RPV was modeled as a separate volume in CORRAL, and thus temporary holdup of the fission products released during the melt phase was calculated. These fission products were released to the drywell after vessel failure. This was not the case in the current calculation which assumed no primary system holdup.

Another important aspect of the model relates to pool decontamination factors. In CORRAL, if flow between compartments goes via the suppression pool, the effect of pool scrubbing can be calculated directly by subjecting the flowing fission products to an appropriate pool decontamination factor. However, as the primary system is not modeled as a volume in the CORRAL model, fission products released during the Gap and Melt stages have to be input directly into appropriate containment volumes. For LOCAs, the release is directly to the drywell airspace so that the core release fraction in Table 4.1 can be used directly. However, for transients, the release is via the SRVs through the suppression pool and into the wetwell airspace. Thus, the Gap and Melt releases may be subject to pool scrubbing. This pool scrubbing is modeled in the CORRAL model by simply dividing the core release fractions in Table 4.1 by the appropriate pool DF.

Each of the ten damage states identified in Section 2 have been analyzed. For two damage states (I-T and IV-T) three potential containment building failure locations (DW, WW and $\overline{WW}$) were considered while in the remaining sequences, only one failure mode was treated. A discussion for each CORRAL-II calculation follows.

4.1 Class I (Damage States I-T and I-S)

Since this accident class has a relatively high frequency of occurrence, a complete series of calculations was carried out for all three failure modes. The LGS-PRA and the previous BNL analysis only considered the $C_1\gamma$ failure mode (equivalent to I-T/DW in the present analysis) because this failure mode results in the largest release of fission products. Since the RPV fails prior to the containment failure, and since the containment is inerted, it is assumed that no oxidation release occurs in this class. This assumption is consistent with the base case technology. The calculated release fractions for each failure mode will now be discussed separately assuming that the accident is initiated by a transient event (damage state I-T). Release fractions based on LOCA initiated sequences are discussed in Section 4.1.4.

4.1.1 Failure in Drywell (DW)

In this sequence, the containment failure is assumed to occur in the drywell wall. This implies that any activity airborne in the drywell atmosphere at the time of failure can enter the environment without first passing through the suppression pool.

Thus, nuclides which are emitted during the vaporization release phase, and which are not carried down into the suppression pool, or agglomerate and settle on the drywell floor can be expected to be released. An inspection of Table 4.2 indicates that the release fraction of Te, Ru, and La are approximately 8.4%, 6.2%, and 7.6% of the possible release, respectively. For I, Cs, and Ba, these values are .18%, 1.8%, and .9%, respectively. This is because the former three groups are released primarily during the vaporization release (particularly Te) and the latter three are released primarily during the gap and melt release phases and thus are subject to a pool DF, which is 100 in this sequence.

A comparison between the current analyses and the LGS-PRA and NUREG/CR-3028 analysis is also shown in Table 4.2. In this case, the suppression pool DF for all cases is 100. The two BNL calculations predict approximately the same fission product release fractions for those nuclides released during the gap and melt release phase. The nuclides emitted during the vaporization release phase are predicted to be higher in the current calculation. The difference in the latter case can be attributed, partially, to an improved thermal-hydraulic representation in the current calculation. The inclusion of a flash release at the time of containment failure is the primary reason why there is an increased release for I_2, Cs, and Ba for the LGS-PRA. Other modeling differences such as temporary (until vessel failure) primary system holdup and different thermal-hydraulic representation account for the remaining differences.

4.1.2 Failure in the Wetwell (WW)

In this case the containment failure occurs in the wetwell airspace. Thus any fission products which enter the outside environment have to pass through the suppression pool. No distinction is made in the DF to which the aerosols are subjected, whether they are released through the SRVs at the base of the suppression pool or through the downcomers at a shallower level. From Table 4.2, it can be seen that in this case the release fractions are substantially lower than in the I-T/DW release path. This is due to the above-mentioned reasons, and in addition, that the wetwell airspace is also available for particle agglomeration and settling.

4.1.3 Failure in the Wetwell with Loss of Suppression Pool ($\overline{WW}$)

In this case the suppression pool is assumed to drain away at the time of containment failure. Thus, although the melt and gap release fractions are fully scrubbed, that portion of the vaporization release, which does not flow down into the wetwell via the suppression pool before containment failure will not be scrubbed. This portion of the vaporization release is assumed to be airborne in the drywell at the time of failure and will thus flow down into the wetwell, where it will be subject to settling. Thus agglomeration and settling is the only attenuation mechanism acting on this portion of the vaporization release. Table 4.2 shows that the release fractions for the I-T/$\overline{WW}$ failure mode are approximately twice as large as these for the I-T/WW failure mode but still substantially below those of the I-T/DW release path.

4.1.4 Class 1 Sequences Initiated by LOCAs (I-S Damage State)

This sequence has been described above. However, two differences have to be accounted for to model a LOCA initiating sequence. First, a small break LOCA is assumed to initiate the accident sequence, rather than a transient event.

Second, since the primary system is open from the start of the transient, the gap and melt release is not through the SRV's, into the suppression pool and from there into the wetwell airspace, but rather directly into the drywell airspace. Thus, in this sequence, only a portion of the gap and melt release is subject to suppression pool scrubbing. Only those fission products that are swept from the drywell into the suppression pool via the downcomers are subjected to pool scrubbing. The remainder of the gap and melt release will be subject to attenuation by the process of agglomeration and settling. This removal mechanism has a substantial affect on the aerosol fission product groups. Elemental iodine will be subject to plate out on surfaces.

Table 4.10 shows a comparison of a transient initiated I-T/DW failure mode and a LOCA initiated I-S/DW failure mode. It is seen that the noble gases and organic iodine release fraction are essentially identical. The elemental iodine release fraction is higher in the case of the LOCA scenario. This indicates that the pool scrubbing, assumed in the transient event, is more efficient than the plate out removal mechanisms which dominates the LOCA event. A comparison of the aerosol fission product groups shows that the fractions released are lower for the LOCA event. In this case the agglomeration and settling of the gap and melt release outweighs the suppression pool scrubbing of the transient event. The time available for agglomeration and settling in this sequence is at least 3.5 hrs.

4.2 Class II (Damage State II-T)

Class II sequences are characterized by long-term overpressurization of the containment building due to steam generation. The RPV failure occurs after the containment has failed and (since the pool is saturated) the DF is 1 in the current "base case" calculation. In NUREG/CR-3028, a DF of 1 was assumed for the aerosol release fractions (Cs-La) and a DF of 10 was assumed for elemental iodine. In the LGS-PRA, a DF of 10 was assumed for all releases, except for the noble gases and organic iodine.

This release is characterized by a rather small containment failure area. Thus, the blowdown to ambient pressure is slow. In view of the slow depressurization in this sequence, it was not clear whether sufficient oxygen would enter into the containment building atmosphere to ensure an oxidation release at the time of RPV failure. This failure occurs approximately 7 hours after the containment has failed. Thus, the fission product transport calculations were carried out with and without the oxidation release.

A direct comparison between the current BNL calculation and the other calculations (NUREG/CR-3028 and LGS-PRA) is difficult since in the current calculation, a failure in the wetwell airspace is assumed, while in the other two calculations, a failure in the drywell was assumed. The largest difference between these two sets of calculations occurs for elemental iodine. The bulk of this difference is directly attributible to the change in pool DF from 10 to 1. Furthermore, it will be noted from Table 4.3 that the species, emitted primarily during the melt release phase (Cs-Ba), are higher in the current calculations, whereas the species emitted during the vaporization phase (Te, Ru, and La) are lower in the current calculation.

This difference is partially due to the location of the failure. In the current calculation, the failure is in the wetwell above the suppression pool and thus nuclides release to the wetwell airspace (melt and gap release) escape directly to the environment at the time of containment failure. In the case of the vaporization release, the release path is not as direct, since the vaporization release will be airborne in the drywell. The drywell atmosphere has to pass through the downcomers to the wetwell airspace and then escapes to the environment. It is then subject to attenuation due to settling in both volumes, which tends to reduce the vaporization release for the current calculation. Only one failure location was considered in the current calculation because attenuation due to the suppression pool has been eliminated (DF=1), and thus it was felt that the release fraction would not be a strong function of the failure location. This assumption will be discussed further in Section 4.4.

A comparison between the release fractions with and without oxidation release in Table 4.3 shows very little change, except for the Ru release, which is almost doubled if an oxidation release is assumed. This release is a direct reflection of the "base case technology" assumptions.

4.3 Class III (Damage State III-T)

This is an ATWS accident in which the containment fails a little after the time of RPV failure, due to steam pressurization. Thus the suppression pool is saturated throughout all of the fission product release periods and consequently the suppression pool DF is 1 for all fission products, except noble gases and organic iodine. Since the containment is still intact at the time of RPV failure, no oxidation release was assumed. In the previous BNL calculation, it was assumed that the DF was 10 for elemental iodine and unity for all aerosol groups. The LGS-PRA assumed a DF of 10 for all fission product groups except the noble gas and organic iodine. Since in the current BNL calculation the vaporization release phase starts approximately 36 minutes before containment failure and then proceeds for 1.4 hrs with a failed containment. The nuclides released during this phase are again prominant contributors to the fission product release (Te, Ru, and La). Results for these calculations are shown in Table 4.4. Only one failure location was considered in the current calculation becaus attenuation due to the suppression pool does not exist (DF=1), and thus it was felt that the release fractions would not be a strong function of the failure location. This dependency will be discussed in Section 4.4.

Comparison between the current calculation and NUREG/CR-3028 and the LGS-PRA are difficult since in this case, a failure in the wetwell above the suppression pool was assumed. The previous calculations (LGS-PRA and NUREG/CR-3028) assumed a failure in the drywell. An inspection of Table 4.4 indicates that in all cases the current BNL predictions are substantially higher than the two previous calculations. This is partially due to the different DF's used and partially due to the different failure modes assumed. This latter difference allows for more settling and plate out in the current calculation for those nuclides released during the vaporization release phase (Te, Ru, and La) but less for the melt and gap release. However, the greatest difference between these calculations, other than the pool DF's for the LGS-PRA, is the timing.

From Table 4.4 it is seen that in the current calculation, the vaporization release phase starts at 2 hrs and ends at 4 hrs. The containment building fails at 2.67 hrs, this allows for approximately 1.4 hrs (or 70%) of the vaporization release to be emitted into an open containment. An inspection of the timing for the previous BNL calculation shows that approximately 36 minutes (or only 30%) of the vaporization release is emitted into an open containment, and in the LGS-PRA all the release is emitted into a closed containment. Therefore, one would expect that the opportunity for fission products to leak into the environment would be largest for the current BNL calculation. This is borne out by the results, and this effect is greater than the influence of the failure location in this sequence.

4.4 Class IV Sequences

For this accident class, sequences initiated by transients and LOCAs were considered and are discussed separately in the following sections.

4.4.1 Class IV Transients (Damage State IV-T)

Since this accident sequence is a major contributor to risk at the LGS, all three failure modes were analyzed separately. Furthermore, this sequence represents an ATWS sequence in which the power is maintained at 30% of rated power by coolant injection. This results in rapid pressurization and leads to containment failure in approximately 40 minutes. Since the suppression pool is saturated, the DF is assumed to be 1 for all fission product groups except for the noble gases and organic iodine. The LGS-PRA used a value fo 10 for the DF, except when the containment failure occurred in the wetwell below the suppression pool where a DF of 1 was used, since the pool was assumed to have drained away before fission product release. In NUREG/CR-3028, a DF of 10 was used for elemental iodine and 1 for all the aerosol species. The exception again being the wetwell failure location when a DF of 1 was used because of loss of the suppression pool.

The containment failure time in the sequence is early, and the blowdown essentially complete by the time the core starts to melt and the release of fission products commences. Thus, the release fractions for all three failure modes in the current BNL calculations are expected to be of similar magnitude. The presence or absence of the suppression pool plays no role (because the pool DF = 1) except to change the airspace in the wetwell. By comparing the three current BNL calculations shown on Tables 4.5, 4.6, and 4.7, it is seen that the release fractions are indeed quite close.

For the $C_4\gamma$ (equivalent to the current IV-T/DW failure mode) which was the only failure location rigorously analyzed in NUREG/CR-3028, the release fractions are slighly higher for the aerosols. This is primarily due to the different thermohydraulic representation used. The large difference for elemental iodine is due to the different DF used. A comparison of the $C_4\gamma''$ release, shown in Table 4.7, in which a DF of 1 was used for all cases, shows similar results for all cases. There is particularly good agreement between the LGS-PRA release fraction and those predicted by the current BNL calculation ($\text{IV-T}/\overline{\text{WW}}$ failure mode).

4.4.2 Class IV LOCAs (Damage State IV-A)

This sequence is identical to the Class IV sequences outlined in Section 4.4 with the exception that a large LOCA event is additionally imposed at the start of the accident. The different location of the melt and gap release (drywell, rather than the wetwell) has a small effect in this case since the pool DF is 1 (saturated pool). The difference between these two calculations is that in the transient, the melt and gap release to the environment is controlled by flow from the wetwell back into the drywell and then out, while in the LOCA case, the release is direct, and only controlled by the flow out of the rupture. However, since the suppression pool is saturated in this case, it has no effect on the fission product release fractions.

Table 4.11 shows a comparison between the transient and the LOCA release fractions. It is seen that they are very similar, with the LOCA only slightly higher. This difference can be ascribed to the longer path required by the melt and gap release in the transient case.

4.4.3 Class IV Reanalysis

The calculational procedure described in Sections 4.4.1 and 4.4.2 results in very little fission product retention and extremely high release to the environment. After review of a draft version of this report we were requested by NRC staff to revise the Class IV calculations. These revised calculations are described in this section. The containment response calculations described in Section 3.5 were used as input to the revised fission product transport calculations. The reanalyzed fission product releases are shown on Tables 4.5, 4.6, 4.7, and 4.11.

In the revised calculations the in-vessel release was divided into two phases. The first phase involves 70% of the release and was released via the SRVs to the wetwell volume. The second phase (involving 30%) is added to the oxidation release and is thus released as a puff at the time of primary system failure. This splitting of the in-vessel release is consistent with the analysis carried out in WASH-1400. However, this division of the in-vessel release only applies to the transient sequences. For the sequence initiated by a large break LOCA, all of the in-vessel release was assumed to be released directly to the drywell. In addition, we also added a third volume to the analysis of fission product transport. This volume represents the reactor building, which was neglected in the analyses described in Sections 4.4.1 and 4.4.2. The inclusion of a reactor building volume in this analysis was also made consistent with the approach taken in WASH-1400.

Both of the above mentioned revisions increase the retention of fission products. In the first assumption, part of the enhanced oxidation release is passed into the wetwell at the time of vessel failure. It is thus subject to agglomeration and settling in both the wetwell and drywell. The flow into the reactor building during this phase of the accident is choked and thus pressure changes in the drywell due to vessel failure have only a slight effect on the flow rate. The addition of a third volume enhances the fission product retention by increasing the volume available for agglomeration and settling before the fission products leak into the environment. From the above discussion it can be concluded that if these methods were also applied to any of the other sequences, the release fractions would be reduced. However, for the other accident sequences, the suppression pool is subcooled and the containment fails late so that fission product attenuation is dominated by these mechanisms. Consequently, it was not considered necessary to also revise the fission product release calculations for the other sequences.

The results of this re-evaluation are shown on the last column of Tables 4.5, 4.6, 4.7, and 4.11. It is seen that for noble gases and organic iodine, there is essentially no change in the fraction released. However, for elemental iodine and the particulate species, the release fractions have been reduced by a factor of approximately 1.5 to 2. The timing of the release is not appreciably affected by these changes.

4.5 Class IS Sequences

These sequences are based on the LGS-SARA[7] and on the descriptions of the accidents in Section 2 and the MARCH analyses in Section 3.5. Two sequences were modeled, namely, TSRB and TSRBCM. The major difference relates to the failure to scram for the TSRBCM sequence. In these cases the suppression pool is subcooled, thus the DF is 100, except for noble gases and organic iodine. For these sequences, the RHR suction lines are assumed to fail at the start of the accident. Failure of the RHR suction lines results in partial draining of the suppression pool, which leaves the SRV submerged but exposes the downcomers. Thus, for transients, the gap and melt releases are scrubbed by the pool but the oxidation and vaporization releases do not pass through the pool. By inspection of the release fractions in Table 4.8 it is seen that those fission product groups with a high release during the vaporization or oxidation release phase (Te, Ru, and La) are major contributors to the release fractions. The addition of an oxidation release at the time of RPV failure, into an open containment enhances the Ru release even more. Those fission product groups which are released primarily during the melt release (I, Cs, and Ba) are quite low, especially Ba, which is essentially only released during the melt release phase. Thus, the release fractions for these sequences are essentially proportional to their vaporization release fraction, except Ru, which is enhanced by an oxidation release, and the noble gases and organic iodine both of which are entirely released.

4.6 Class S Sequences

For this class the containment and the vessel fail at the start of the accident. Thus, all fission product release bypasses the suppression pool. However, although the various releases take place into a failed containment building with no suppression pool DF, it is also evident that the flow rate out of the building at the time of release will be comparatively low. Thus, the fission products in the aerosol group are subject to attenuation by agglomeration and settling. An inspection of Table 4.9 indicates that for the

S-$\overline{H2O}$ sequences for those species dominated by melt release (I, Cs, and Ba) approximately 25% of the total release fraction escapes to the environment, while for those nuclides released primarily during the vaporization release phase (Te and La) approximately 38% are released to the environment. The Ru release fraction is enhanced by the oxidation release at the point of RPV failure. It is seen that the release fractions for the S-$\overline{\text{H2O}}$ sequences are slightly higher than for the S-H2O sequences. This is particularly due to the delayed start of the vaporization release phase in the S-H2O case.

In the latter sequence the vaporization release starts approximately one hour later. Since the flow out of the containment building is dropping off with time, the leakage to the environment from the vaporization release becomes a smaller contributor to the overall release.

A comparison between the BNL release fractions and the LGS-SARA release fractions for the VR sequence (equivalent to S-$\overline{\text{H2O}}$/WW) shows good agreement. The only difference being due to the use in the LGS-SARA of fission product release coefficients based on NUREG-0772.[8] Thus, the organic iodine is lower (approximately a factor 22) and the barium release is higher. A similar comparison for the VRH2O sequence (equivalent to S-H2O/WW) does not show the same level of agreement. This difference can only be attributed to the large release of fission products during the melt release in the LGS-SARA because the NUREG-0772 fission product release coefficients are used. The large melt release is assumed to be airborne in the RPV and is expelled at the time of core slump. In the BNL approach, this release fraction is lower. Furthermore, the release is deposited in the containment building and not held up in the RPV since the latter is not explicitly modeled. These differences in fission product transport and thermal-hydraulic modeling account for the lower release fraction in the BNL case.

4.7 Summary

In this section the fission product release fractions and the associated timing is presented. These determinations are based on the base case technology as outlined above, and the release fractions are summarized in Tables 4.2-4.9. The time of release is defined as the time of containment failure for those cases in which the meltdown takes place in an intact containment building. For those cases, when the containment building fails prior to core damage, the time of releases is defined as the start of core melting. The duration of release will be defined as the time for the containment building to blow down to atmospheric pressure. However, if the building fails first (meltdown into a failed containment building) the duration of release will be from the start of core melting to the completion of the vaporization release. The warning time is defined as the time period between the start of core melt and the time of containment failure. If the containment building fails first, the warning time is defined as the difference between the start of core melt and the time of containment failure.

The energy of release is the energy release rate associated with the plume at the time of failure. This value is extracted from the MARCH calculation (refer to Section 3). In those cases where the release is spread out over many hours, the energy of release is very low. The height of release is chosen to be 25 m (82 feet) in all cases. The information in Tables 4.2

through 4.9 is used in the following section to generate the source terms for used in the DES for the LGS.

4.8 References to Section 4

1) NRC Memorandum from B. Sheron, Branch Chief/RSB to R. Mattson, Director, DSI," Proposed PRA Methodology for Limerick and GESSAR," dated July 22, 1983.

2) Reactor Safety Study, "An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG/75-014, 1975.

3) "Radionuclide Release Under Specific LWR Accident Conditions, Volume 2: BWR, Mark I Design and Volume 3: BWR, Mark III Design," BMI-2104 draft report.

4) R. J. Burian and P. Cybulskis, "CORRAL 2 User's Manual," BCL report dated January 1977.

5) R. O. Wooton and H. I. Avci, "MARCH Code Description and User's Manual," NUREG/CR-1711, October 1980.

6) Philadelphia Electric Company, "Limerick Generating Station, Probabilistic Risk Assessment," March 1981.

7) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," April 1983.

8) "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," USNRC Report NUREG-0772, June 1981.

Table 4.1 Fission Product Release Source Summary - Best Estimate Total Core Release Fractions.

Fission Product	Gap Release Fraction	Meltdown Release Fraction	Vaporization Release Fraction[(d)]	Steam Explosion Fraction[(e)]
Xe, Kr	0.030	0.870	0.100	(X)(Y) 0.90
I, Br	0.017	0.883	0.100	(X)(Y) 0.90
Cs, Rb	0.050	0.760	0.190	--
Te[(a)]	0.0001	0.150	0.850	(X)(Y) (0.60)
Sr, Ba	0.000001	0.100	0.010	--
Ru[(b)]	--	0.030	0.050	(X)(Y) (0.90)
La[(c)]	--	0.003	0.010	--

(a) Includes Se, Sb

(b) Includes Mo, Pd, Rh, Tc

(c) Includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

(d) Exponential loss over 2 hours with halftime of 30 minutes. If a steam explosion occurs prior to this, only the core fraction not involved in the steam explosion can experience vaporization.

(e) X = Fraction of core involved in the steam explosion. Y = Fraction of inventory remaining for release by oxidation.

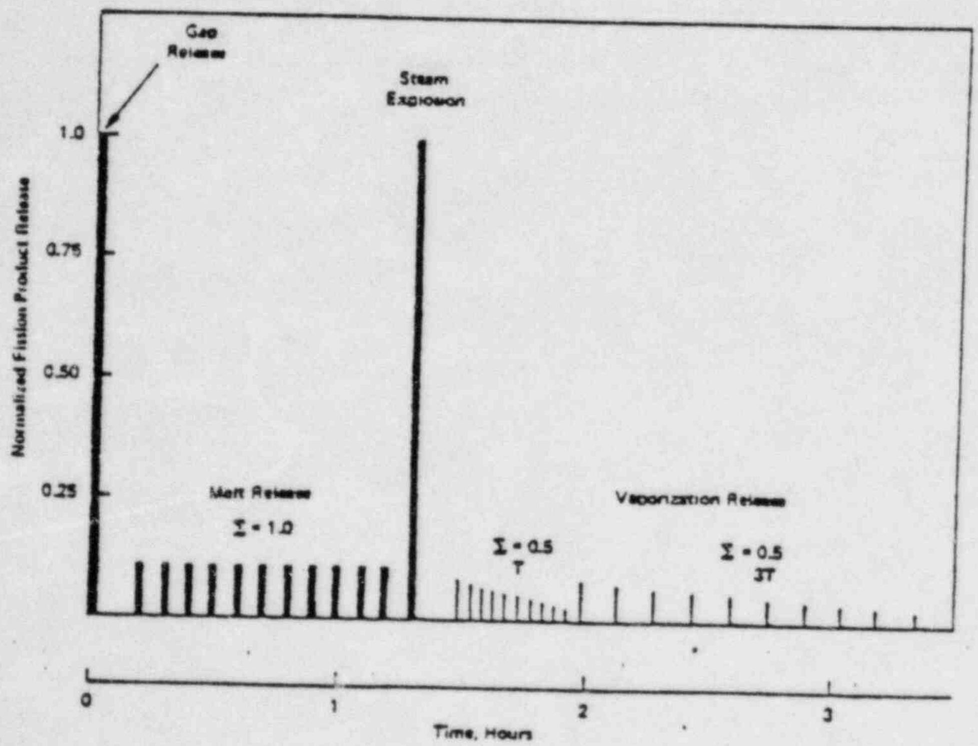

Figure 4.1 Typical Sequence of Spike Fission Product Releases for Postulated Accidents.

Table 4.2 Fission product release fractions for Class 1

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION		
FAILURE MODE	$C_1\gamma$	$C_1\gamma$	I-T/DW	I-T/WW	I-T/$\overline{WW}$
OXIDATION RELEASE	Yes	Yes	No	No	No
Xe - Kr	1.0	.939(-1)	1.0	1.0	1.0
Organic Iodine	---	---	6.99(-3)	6.99(-3)	6.99(-3)
I_2	1.1(-1)	9.3(-3)	1.78(-3)	1.48(-4)	2.09(-4)
Cs	9(-2)	2.0(-2)	1.88(-2)	3.11(-4)	9.19(-4)
Te	1.6(-2)	4.6(-2)	8.41(-2)	1.23(-3)	2.16(-3)
Ba	1.0(-2)	1.7(-3)	9.94(-4)	1.91(-5)	8.22(-5)
Ru	3.0(-3)	3.0(-3)	4.95(-3)	7.39(-5)	1.39(-4)
La	3.0(-4)	6.1(-4)	9.89(-4)	1.46(-5)	2.61(-5)
DF for I_2	100	100	100	100	100
DF for Aerosols	100	100	100	100	100
Core Melt Start	1.3	1.75	1.5	1.5	1.5
Core Melt End	2.5	2.43	2.42	2.42	2.42
1st Vap. Release			2.90	2.90	2.90
2nd Vap. Release			3.40	3.40	3.40
Vap. Release End			4.90	4.90	4.90
Containment Fail	6.5	5.23	5.17	5.17	5.17

Table 4.3 Fission product release fractions for Class II

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	
FAILURE MODE	$C_2\gamma$	$C_2\gamma$	II-T/WW	II-T/WW
OXIDATION RELEASE	Yes	Yes	Yes	No
Xe - Kr	1.0	1.0	9.8(-1)	9.8(-1)
Organic Iodine	---	---	6.86(-3)	6.86(-3)
I_2	6(-2)	1.56(-1)	6.73(-1)	6.16(-1)
Cs	2.3(-2)	2.58(-1)	3.36(-1)	3.36(-1)
Te	4.0(-1)	4.21(-1)	2.31(-1)	2.38(-1)
Ba	6.3(-3)	2.7(-2)	4.1(-2)	4.1(-2)
Ru	6.9(-2)	7.0(-2)	4.0(-2)	2.2(-2)
La	4.7(-3)	5.4(-3)	3.3(-3)	3.3(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	36.6	35.5	24.92	24.92
Core Melt End	39.0	38.3	26.83	26.83
1st Vap. Release			26.83	26.83
2nd Vap. Release			27.33	27.33
Vap. Release End			28.83	28.83
Containment Fail	30.0	29.2	19.6	19.6

Table 4.4 Fission product release fractions for Class III

ASSESSMENT	LGS - [illegible]A	NUREG-3028	DES CALCULATION
FAILURE MODE	$C_3\gamma$	$C_3\gamma$	III-T/WW
OXIDATION RELEASE	Yes	Yes	No
Xe - Kr	1.0	1.0	9.99(-1)
Organic Iodine	---	---	6.99(-1)
I_2	4.02(-2)	1.22(-1)	7.8(-2)
Cs	2.4(-2)	5.42(-2)	2.24(-1)
Te	7.3(-2)	1.85(-1)	5.74(-1)
Ba	2.7(-3)	3.61(-3)	1.95(-2)
Ru	8.6(-3)	1.7(-2)	3.65(-2)
La	9.1(-4)	2.4(-3)	6.92(-3)
DF for I_2	10	10	1
DF for Aerosols	10	1	1
Core Melt Start	.85	.76	.5
Core Melt End	2.5	2.22	1.8
1st Vap. Release			2.05
2nd Vap. Release			2.55
Vap. Release End			4.05
Containment Fail	6.5	3.83	2.67

Table 4.5 Fission product release fractions for Class IV (failure location DW)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma$	$C_4\gamma$	IV-T/DW	IV-T/DW
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	9.99(-1)	9.99(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	2.61(-1)	1.54(-1)	9.39(-1)	4.74(-1)
Cs	2.02(-1)	7.49(-1)	8.61(-1)	4.86(-1)
Te	4.34(-1)	7.47(-1)	8.62(-1)	5.09(-1)
Ba	2.90(-2)	8.60(-2)	9.40(-2)	5.54(-2)
Ru	9.50(-2)	1.10(-1)	1.49(-1)	8.55(-2)
La	5.20(-3)	1.03(-2)	1.15(-2)	6.82(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.20	2.20
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

Table 4.6 Fission product release fractions for Class IV (failure location WW)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma'$	$C_4\gamma'$	IV-T/WW	IV-T/WW
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	1.0	9.99(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	7.0(-2)	9.80(-2)	9.39(-1)	4.61(-1)
Cs	9.0(-2)	7.49(-1)	7.72(-1)	4.81(-1)
Te	2.0(-1)	7.47(-1)	6.88(-1)	4.45(-1)
Ba	1.6(-2)	8.60(-2)	9.0(-2)	5.60(-2)
Ru	8.8(-2)	1.10(-1)	1.19(-1)	7.81(-2)
La	6.0(-3)	1.03(-2)	9.40(-3)	6.03(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.2	2.2
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

Table 4.7 Fission product release fractions for Class IV (failure location $\overline{WW}$ below wetwell waterline)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma''$	$C_4\gamma''$	IV-T/$\overline{WW}$	IV-T/$\overline{WW}$
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	1.0	9.98(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	7.30(-1)	7.08(-1)	8.74(-1)	4.68(-1)
Cs	7.0(-1)	7.49(-1)	8.04(-1)	5.18(-1)
Te	5.50(-1)	7.47(-1)	5.82(-1)	4.81(-1)
Ba	9.0(-2)	8.60(-2)	9.60(-2)	5.96(-2)
Ru	1.20(-1)	1.10(-1)	1.38(-1)	8.31(-2)
La	7.0(-3)	1.03(-2)	7.90(-3)	6.51(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.2	2.2
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

Table 4.8 Fission product release fractions for Class IS

ASSESSMENT	LGS-SARA	DES CALCULATIONS	
FAILURE MODE	TSRB	IS-$\overline{C}$/DW	IS-C/DW
OXIDATION RELEASE	-	Yes	Yes
Xe - Kr	1.0	9.99(-1)	9.99(-1)
Organic Iodine	3.0(-4)	6.99(-3)	6.99(-3)
I_2	5.0(-2)	8.2(-2)	7.6(-2)
Cs	9.0(-2)	1.43(-1)	1.37(-1)
Te	9.0(-2)	6.06(-1)	5.68(-1)
Ba	4.0(-3)	7.78(-3)	7.42(-3)
Ru	2.0(-2)	1.07(-1)	8.2(-2)
La	5.0(-3)	7.37(-3)	7.05(-3)
DF for I_2		100	100
DF for Aerosols		100	100
Core Melt Start		1.47	.37
Core Melt End		2.32	1.28
1st Vap. Release		2.37	1.53
2nd Vap. Release		2.87	2.03
Vap. Release End		4.37	3.53
Containment Fail		0.0	0.0

Table 4.9 Fission product release fractions for Class S

ASSESSMENT	LGS-SARA		DES CALCULATIONS	
FAILURE MODE	VRH2O	VR	$S\text{-}H2O/\overline{WW}$	$S\text{-}\overline{H2O}/\overline{WW}$
OXIDATION RELEASE			Yes	Yes
Xe - Kr	1.0	1.0	9.87(-1)	9.68(-1)
Organic Iodine	3.0(-4)	3.0(-4)	6.99(-3)	6.98(-3)
I_2	5.0(-1)	1.0(-1)	1.09(-1)	2.55(-1)
Cs	7.3(-1)	3.3(-1)	1.62(-1)	2.74(-1)
Te	7.5(-1)	3.3(-1)	2.90(-1)	3.86(-1)
Ba	3.5(-1)	1.5(-1)	1.20(-2)	2.60(-2)
Ru	7.0(-2)	4.0(-2)	4.90(-2)	6.20(-2)
La	5.0(-2)	2.0(-2)	3.64(-3)	4.99(-3)
DF for I_2			1	1
DF for Aerosols			1	1
Core Melt Start			2.67	2.83
Core Melt End			3.65	3.85
1st Vap. Release			5.23	4.38
2nd Vap. Release			5.73	4.88
Vap. Release End			7.23	6.38
Containment Fail			0.0	0.0

Table 4.10 A comparison of fission product release fractions for Class I sequences initiated by LOCAs and Transients

ASSESSMENT	DES CALCULATION	
FAILURE MODE	I-S/DW	I-T/DW
OXIDATION RELEASE	No	No
Xe- Kr	9.99(-1)	9.99(-1)
Organic Iodine	6.99(-3)	6.99(-3)
I_2	3.31(-3)	1.78(-3)
Cs	4.89(-3)	1.88(-2)
Te	2.80(-3)	8.41(-2)
Ba	6.01(-4)	9.94(-4)
Ru	2.87(-4)	4.95(-3)
La	4.01(-4)	9.89(-4)
DF for I_2	100	100
DF for Aerosols	100	100
Core Melt Start	1.35	1.5
Core Melt End	2.44	2.42
1st Vap. Release	2.83	2.90
2nd Vap. Release	3.33	3.40
Vap. Release End	4.83	4.90
Containment Fail	5.11	5.17

Table 4.11 A comparison of fission product release fractions for Class IV sequences initiated by LOCAs and Transients

ASSESSMENT	NUREG-3028	DES CALCULATIONS		FES CALCULATIONS
FAILURE MODE	$C_4{}^{\gamma}$ LOCA	IV-A/DW	IV-T/DW	IV-A/DW
Xe - Kr	1.0	.9989	.999	9.96(-1)
Organic Iodine	7.0(-3)	6.99(-3)	6.99(-3)	6.94(-3)
I_2	8.23(-1)	9.68(-1)	9.39(-1)	4.78(-1)
Cs	7.50(-1)	8.70(-1)	8.61(-1)	5.06(-1)
Te	7.5(-1)	8.74(-1)	8.62(-1)	5.18(-1)
Ba	8.6(-2)	9.94(-2)	9.39(-2)	5.76(-2)
Ru	1.11(-1)	1.5(-1)	1.49(-1)	8.86(-2)
La	1.0(-2)	1.17(-2)	1.15(-2)	6.95(-3)
DF for I_2		1	1	1
DF for Aerosols		1	1	1
Core Melt Start		1.17	1.13	1.13
Core Melt End		1.58	2.20	2.20
1st Vap. Release		2.20	2.47	2.47
2nd Vap. Release		2.70	2.97	2.97
Vap. Release End		4.20	4.47	4.47
Containment Fail		.67	.67	.67

5.0 SOURCE TERM CHARACTERISTICS

In this section we generate representative source terms for the various failure modes and release paths. This section therefore assembles the information contained in Sections 2 through 4 of this report. The probabilities of the failure modes were calculated in Section 2 and rely on information obtained from the LGS-PRA,[1] the LGS-SARA,[2] the BNL reviews[3,4] of these reports. The timing of fission product release, energy of release, duration of release and warning time for the various failure modes were based on the MARCH analysis in Section 3. The quantities of the fission products released were calculated in Section 4. Source terms for 27 failure modes and release paths have been determined. Fourteen of these source terms were calculated as part of the present study and are described in detail in the body of this report. The remaining thirteen source terms are based on the information in References [1-3] and on the Reactor Safety Study[5] (with modifications to reflect the present assessment).

The information contained in this section is the data needed to perform a site consequence analysis. The Accident Evaluation Branch (AEB) at NRC has the responsibility of performing the site consequence analysis for the Limerick site as part of the Draft Environmental Statement (DES) and Final Environmental Statement (FES). The information in Tables 5.1 through 5.7 was generated specifically as input to the DES. The information in Tables 5.6 through 5.12 was used as input to the FES. In the following sections we will briefly summarize the source terms.

5.1 LGS-DES Source Terms

5.1.1 Source Terms for Damage State I-T (Table 5.1)

This damage state is defined in Section 2 and basically consists of transients with loss-of-inventory make-up. Core melt is relatively fast and occurs into an intact containment. After vessel failure the majority of the core materials are retained on the diaphragm floor. Containment failure occurs via gradual overpressurization (except for SE, HB, LGT, and $\overline{\text{LGT}}$ releases) several hours after vessel failure due to core/concrete interactions. Each of the source terms in Table 6 are discussed below.

I-T/DW

This release path assumes a failure in the drywell wall. The gap and melt releases are directed to the suppression pool and subjected to a DF of 100 (because the water is subcooled) before reaching the wetwell airspace. The vaporization release is directed to the drywell without any pool scrubbing. All fission products in the drywell and wetwell are subjected to agglomeration and settling as predicted by CORRAL[6] prior to vessel failure several hours after the pressure vessel failure.

I-T/WW

This release path assumes a failure in the wetwell above the suppression pool. The gap, melt, and vaporization releases are released to the drywell and wetwell as described above. The only difference is that at containment failure fission products in the drywell must pass through the downcomers and suppression pool prior to release to atmosphere.

I-T/$\overline{WW}$

This release path assumes a failure in the wetwell below the suppression pool, which drains the water. The gap, melt, and vaporization releases are again released to the containment as described above. The only difference is that at containment failure the suppression pool is drained so that fission products in the drywell no longer have to pass through the suppression pool (as in the I-T/WW release path) prior to release to atmosphere.

I-T/SE

This release path results from an in-vessel steam explosion generated missile. We assume this occurs at core slump and opens a direct path from the primary system to atmosphere. In the LGS-PRA, this failure mode was similar to the RSS release category BW1. This release corresponds to an ATWS sequence analyzed in Appendix V of the RSS, in which the steam explosion was assumed to occur after only 13% of the core had melted. Consequently, most of the melt release was released to containment without pool scrubbing. This is not consistent with our analysis of this sequence as we would subject all of the melt release to pool scrubbing. We have therefore used a steam explosion release from the RSS that more appropriately reflects our analysis of the sequence.

I-T/HB

This release path results from H_2 burn failures during the time when the containment atmosphere is deinerted. We used the same release category as in the LGS-PRA but reduced the oxidation components associated with the Te and Ru releases. (Note in the LGS-PRA, this release category was representative of ex-vessel steam explosions).

I-T/LGT and $\overline{LGT}$

These release paths results from containment leakage and assume that the SGTS operates (LGT) or that it does not operate ($\overline{LGT}$). We use the LGS-PRA releases but changed the timing to correspond to our analysis.

5.1.2 Source Terms for Damage State II-T (Table 5.2)

This damage state is defined in Section 2 and basically assumes loss of containment heat removal. Eventually, the containment fails, which causes the loss-of-inventory make-up. As the containment has failed prior to core melt and the suppression pool is saturated (DF of 1), the location of containment failure (DW, WW or $\overline{WW}$) is of rather less importance than for the I-T damage states. Each of the source terms in Table 7 are discussed below.

II-T/WW

This release path assumes a failure in the wetwell above the suppression pool. The melt release is directed to the suppression pool but is not subjected to pool decontamination because the water is saturated. The vaporization release is directed to the drywell, then through the downcomers to the wetwell air-space and finally to the atmosphere. This one failure location was also used to represent failures in the drywell (DW) and wetwell below the suppression pool ($\overline{WW}$). This assumption is reasonable because the pool is saturated and hence the different flow paths do not result in significant differences in calculated release fractions (refer to the discussion on the IV-T damage state).

II-T/SE

This release path results from an in-vessel steam explosion generated missile. The release path used in the LGS-PRA, which was taken from Appendix V of the RSS, was considered appropriate and is used in Table 7. Differences related only to the timing, which now corresponds to the present analysis of a II-T damage state.

5.1.3 Source Terms for Damage State III-T (Table 5.3)

This damage state corresponds to a transient event coupled with loss of scram function (refer to Section 2). Core melt is rapid and occurs into an intact containment. Containment failure is predicted to occur after vessel failure due to overpressurization. However, the suppression pool is saturated so that the gap, melt, and vaporization releases are not subjected to decontamination by the pool. Consequently, we again (as for the II-T damage state) used one failure location to represent the three potential locations.

III-T/WW

This release path is similar to the I-T/WW sequence, however (because the pool is saturated) the melt release is not subjected to pool scrubbing in this damage state.

III-T/SE

The steam explosion release category used in the LGS-PRA was considered appropriate and is used in Table 5.3. Differences relate only to timing, which was made consistent with our MARCH analysis.

5.1.4 Source Terms for Damage State IV-T (Table 5.4)

This damage state is defined in Section 2 and essentially consists of ATWS sequences in which continued coolant make-up results in overpressurization failure of containment prior to core melt. The suppression pool is saturated for these sequences and hence the DF is unity. We analyzed the impact of the three potential failure locations (DW, WW and $\overline{\text{WW}}$) and because of the saturated pool found similar release fractions (refer to Table 5.4). These calculations support our use of only one failure location for the II-T and III-T damage states. The release paths for the three locations have been discussed in detail above and will not be repeated here.

IV-T/SE

The steam explosion release category used in the LGS-PRA for Class III (damage state III-T) was considered appropriate to this damage state. Consequently, this release category is used with the timing changed to be consistent with our MARCH analysis.

5.1.5 Source Terms for Damage States I-S and IV-A (Table 5.5)

These damage states are defined in Section 2 and correspond to LOCA initiated sequences. They were calculated to have a low frequency but (because of differences in flow paths relative to transients) were analyzed separately. The I-S/DW flow path results in the release of the melt and vaporization releases to the drywell, thus bypassing pool scrubbing. However, as containment fails several hours after vessel failure, the release fractions are not significantly different from the I-T/DW flow path (in which the gap and melt releases were subjected to suppression pool scrubbing).

5.1.6 Source Terms for Damage States IS-C and IS-$\overline{\text{C}}$ (Table 5.6)

These damage states are defined in Section 2 and are seismically induced. The RHR suction lines are severed resulting in partial loss of the suppression pool. The gap and melt releases are directed to the suppression pool and are subjected to decontamination (the water is subcooled and the DF=100) before release via the severed RHR suction lines. The vaporization release is directed to the drywell and then flows through the downcomers into the wetwell. However, as the suppression pool has drained below the downcomer outlet, the vaporization release is not subjected to pool scrubbing. The difference between IS-C and IS-$\overline{\text{C}}$ relates to the scram function and does not influence the flow paths, only the timing of the sequence is affected.

The in-vessel steam explosion failures (IS-C/SE and IS-$\overline{\text{C}}$/SE) were assumed to be similar to the I-T/SE release. Only the timing was altered to reflect the MARCH analysis.

5.1.7 Source Terms for Damage States S-H20 and S-$\overline{\text{H20}}$ (Table 5.7)

These damage states are defined in Section 2 and are also seismically induced. The RHR suction lines are severed but also the vessel fails at the start of the accident. Thus the core melts into a failed containment and none of the releases are subjected to pool scrubbing. The only differences between the S-H20 and S-$\overline{\text{H20}}$ sequences relates to the location of the failure in the vessel. For the S-H20 sequence, water remains in the vessel and is available for interacting with the core debris as it slumps. This will affect movement of the fission products and also allows the potential for an in-vessel steam explosion. As the melt release is not subject to pool scrubbing, the steam explosion release was considered similar to the release used for release paths III-T/SE and IV-T'SE.

The S-$\overline{H2O}$ damage state involves a failure of the vessel, such the water is completely drained at the start of the accident. Thus, there is no in-vessel debris/water interaction and no potential for an in-vessel steam explosion.

5.2 LGS-FES Source Terms

The source terms for damage states IS-C, IS-$\overline{C}$, S-H2O, and S-$\overline{H2O}$ (in Tables 5.6 and 5.7) were not changed for use in the LGS-FES relative to the LGS-DES. The frequencies of the source terms for damage states I-T, II-T, and III-T were changed to reflect the revised probabilities of sequences initiated by loss-of-offsite-power and fire (refer to Section 2). The revised source term probabilities are given in Tables 5.2, 5.9, and 5.10. In addition, we recalculated the source terms for damage states IV-T and IV-A for input to the LGS-FES (refer to Sections 3.5 and 4.4.3). The revised source terms are given in Tables 5.11 and 5.12.

5.3 References to Section 5

1) Philadelphia Electric Company, "Limerick Generating Station, Probabilistic Risk Assessment," March 1981.

2) Philadelphia Electric Company, "Limerick Generating Station, Severe Accident Risk Assessment," April 1983.

3) I. A. Papazoglou, et al., "Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028, February 1983.

4) M. A. Azarm, et al., "A Preliminary Review of the Limerick Generating Station Severe Accident Risk Assessment, Volume 1: Core Melt Frequency," Draft BNL report dated August 15, 1983.

5) "Reactor Safety Study: An Assessment of Accident Risk in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/104, 1975.

6) R. J. Burian and P. Cybulskis, "CORRAL 2 User's Manual," BCL report, dated January 1977.

Table 5.1 Summary of source terms for damage state I-T for input to LGS-DES

Failure Modes and Release Paths	I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE	I-T/HB	I-T/LGT	I-T/$\overline{LGT}$
Xe-Kr	1.0	1.0	1.0	1.0	1.0	0.73	0.73
OI	6.99(-3)*	5.99(-3)	6.99(-3)	-	-	-	-
I	1.78(-3)	1.48(-4)	2.09(-4)	9.6(-2)	2.0(-1)	2.7(-3)	1.9(-2)
Cs	1.88(-2)	3.11(-4)	9.19(-4)	1.0(-1)	6.0(-2)	9.8(-5)	9.8(-2)
Te	8.41(-2)	1.23(-3)	2.16(-3)	4.0(-1)	1.0(-1)	4.6(-4)	4.6(-2)
Ba	9.94(-4)	1.91(-5)	8.22(-5)	1.0(-2)	7.0(-3)	1.6(-5)	1.6(-3)
Ru	4.95(-3)	7.39(-5)	1.39(-4)	4.0(-1)	8.0(-2)	3.2(-5)	3.2(-3)
La	9.89(-4)	1.46(-5)	2.61(-5)	2.0(-3)	1.0(-5)	5.8(-6)	5.8(-4)
Time of Release (hr)	5.17	5.17	5.17	2.4	2.4	1.5	1.5
Duration of Release (hr)	0.5	0.5	0.5	0.5	0.5	3.4	3.4
Warning Time (hr)	3.67	3.67	3.67	1.0	1.0	0	0
Energy of Release (10^6 Btu/hr)	100	100	100	130	100	1.0	1.0
Height (ft)	82	82	82	82	82	82	82
Probability (Regional Disasters)	5.6(-7)	5.1(-7)	5.7(-8)	2.3(-10)	2.3(-8)	5.0(-7)	6.2(-7)
Probability (Non-Regional Disasters)	2.41(-5)	2.18(-5)	2.44(-6)	9.77(-9)	9.77(-7)	2.17(-5)	2.67(-5)
Total Probability	2.47(-5)	2.23(-5)	2.5(-6)	1.0(-8)	1.0(-6)	2.22(-5)	2.73(-5)

*6.99(-3) = 6.99×10^{-3}

Table 5.2 Summary of source terms for damage state II-T for input to LGS-DES

Failure Modes and Release Paths	II-T/WW	II-T/SE
Xe-Kr	9.8(-1)*	1.0
OI	6.86(-3)	-
I_2	6.73(-1)	9.6(-2)
Cs	3.36(-1)	1.0(-1)
Te	2.31(-1)	4.0(-1)
Ba	4.1(-2)	1.0(-2)
Ru	4.0(-2)	4.0(-1)
La	3.28(-3)	2.0(-3)
Time of Release (hr)	24.92	27
Duration of Release (hr)	3.91	0.5
Warning time (hr)	5.32	7
Energy of Release (10^6 Btu/hr)	1.0	130.0
Height (ft)	82	82
Probability (Regional Disasters)	2.0(-8)	2.0(-12)
Probability (Non-Regional Disasters)	2.04(-6)	2.03(-10)
Total Probability	2.06(-6)	2.05(-10)

*9.8(-1) = 9.8×10^{-1}

Table 5.3 Summary of source terms for damage state III-T for input to LGS-DES

Failure Modes and Release Paths	III-T/WW	III-T/SE	III-T/HB	III-T/LGT	III-T/$\overline{LGT}$
Xe-Kr	1.0	1.0	1.0	7.3(-1)	7.3(-1)
OI	6.99(-3)*	-	-	-	-
I_2	7.81(-2)	4.0(-1)	2.0(-1)	2.7(-3)	1.9(-2)
Cs	2.24(-1)	4.0(-1)	6.0(-2)	9.8(-5)	9.8(-2)
Te	5.74(-1)	5.0(-1)	1.0(-1)	4.6(-4)	4.6(-2)
Ba	1.95(-2)	5.0(-2)	7.0(-3)	1.6(-5)	1.6(-3)
Ru	3.65(-2)	5.0(-1)	8.0(-2)	3.2(-5)	3.2(-3)
La	6.92(-3)	3.0(-3)	1.0(-5)	5.8(-6)	5.8(-4)
Time of Release (hr)	2.67	2.0	2.0	0.5	0.5
Duration of Release (hr)	1.38	0.5	0.5	3.5	3.5
Warning Time (hr)	2.17	1.0	1.0	0	0
Energy of Release (10^6 Btu/hr)	100	130	100	1.0	1.0
Height (ft)	82	82	82	82	82
Probability (Regional Disasters)	3.7(-7)	7.4(-11)	7.4(-9)	1.6(-7)	2.0(-7)
Probability (Non-Regional Disasters)	1.66(-6)	3.4(-10)	3.4(-8)	7.5(-7)	9.2(-7)
Total Probability	2.03(-6)	4.1(-10)	4.1(-8)	9.1(-7)	1.12(-6)

*6.99(-3) = 6.99 x 10^{-3}

Table 5.4 Summary of source terms for damage state IV-T for input to LGS-DES

Failure Modes and Release Paths	IV-T/DW	IV-T/WW	IV-T/$\overline{WW}$	IV-T/SE
Xe-Kr	1.0	1.0	1.0	1.0
OI	6.99(-3)*	6.99(-3)	6.99(-3)	-
I_2	9.39(-1)	9.39(-1)	8.74(-1)	4.0(-1)
Cs	8.61(-1)	7.72(-1)	8.04(-1)	4.0(-1)
Te	8.62(-1)	6.88(-1)	5.82(-1)	5.0(-1)
Ba	9.39(-2)	9.0(-2)	9.55(-2)	5.0(-2)
Ru	1.49(-1)	1.19(-1)	1.38(-1)	5.0(-1)
La	1.15(-2)	9.38(-3)	7.89(-3)	3.0(-3)
Time of Release (hr)	1.13	1.13	1.13	2.0
Duration of Release (hr)	3.34	3.34	3.34	0.5
Warning Time (hr)	0.5	0.5	0.5	1.5
Energy of Release (10^6 Btu/hr)	1.0	1.0	1.0	130
Height (ft)	82	82	82	82
Probability (Regional Disasters)	4.7(-8)	4.27(-8)	4.75(-9)	9.5(-12)
Probability (Non-Regional Disasters)	1.63(-7)	1.46(-7)	1.63(-8)	3.25(-11)
Total Probability	2.1(-7)	1.89(-7)	2.1(-8)	4.2(-11)

*6.99(-3) = 6.99×10^{-3}

Table 5.5 Summary of source terms for damage states I-S and IV-A for input to LGS-DES

Failure Modes and Release Paths	I-S/DW	IV-A/DW
Xe-Kr	9.99(-1)*	9.99(-1)
OI	6.99(-3)	6.99(-3)
I_2	3.31(-3)	9.65(-1)
Cs	4.89(-3)	8.7(-1)
Te	2.80(-3)	8.74(-1)
Ba	6.01(-4)	9.9(-2)
Ru	2.87(-4)	1.51(-1)
La	4.01(-4)	1.2(-2)
Time of Release (hr)	5.11	1.17
Duration of Release (hr)	0.5	3.0
Warning Time (hr)	3.76	0.5
Energy of Release (10^6 Btu/hr)	100	1.0
Height (ft)	82	82
Probability (Regional Disasters)	-	-
Probability (Non-Regional Disasters)	3.76(-8)	5.0(-9)
Total Probability	3.76(-8)	5.0(-9)

*9.99(-1) = 9.99×10^{-1}

Table 5.6 Summary of source terms for damage states IS-C and IS-$\overline{C}$

Failure Modes and Release Paths	IS-C/DW	IS-C/SE	IS-$\overline{C}$/DW	IS-$\overline{C}$/SE
Xe-Kr	1.0	1.0	1.0	1.0
OI	6.99(-3)*	-	6.99(-3)	-
I_2	7.61(-2)	9.6(-2)	8.22(-2)	9.6(-2)
Cs	1.37(-1)	1.0(-1)	1.43(-1)	1.0(-1)
Te	5.68(-1)	4.0(-1)	6.06(-1)	4.0(-1)
Ba	7.42(-3)	1.0(-2)	7.78(-3)	1.0(-2)
Ru	8.17(-2)	4.0(-1)	1.07(-1)	4.0(-1)
La	7.05(-3)	2.0(-3)	7.37(-3)	2.0(-3)
Time of Release (hr)	0.37	1.3	1.47	2.3
Duration of Release (hr)	3.16	0.5	2.9	0.5
Warning Time (hr)	0.37	1.3	1.47	2.3
Energy of Release (10^6 Btu/hr)	1.0	130	1.0	130
Height (ft)	82	82	82	82
Probability (Regional Disasters)	1.3(-7)	1.3(-11)	9.0(-7)	9.0(-11)
Probability (Non-Regional Disasters)	1.4(-8)	1.4(-12)	1.0(-7)	1.0(-11)
Total Probability	1.44(-7)	1.44(-11)	1.0(-6)	1.0(-10)

*6.99(-3) = 6.99×10^{-3}

Table 5.7 Summary of source terms for damage states S-H20 and S-$\overline{H20}$

Failure Modes and Release Paths	S-H20/$\overline{WW}$	S-H20/SE	S-$\overline{H2O}$/$\overline{WW}$
Xe-Kr	9.87(-1)*	1.0	9.68(-1)
OI	6.99(-3)	-	6.98(-3)
I_2	1.09(-1)	4(-1)	2.56(-1)
Cs	1.62(-1)	4(-1)	2.74(-1)
Te	2.89(-1)	5(-1)	3.86(-1)
Ba	1.23(-2)	5(-2)	2.57(-2)
Ru	4.9(-2)	5(-1)	6.18(-2)
La	3.64(-3)	3.0(-3)	4.99(-3)
Time of Release (hr)	2.67	3.5	2.83
Duration of Release (hr)	4.56	0.5	3.55
Warning Time (hr)	2.67	3.5	2.83
Energy of Release (10^6 Btu/hr)	1.0	130.0	1.0
Height (ft)	82	82	82
Probability (Regional Disasters)	4.1(-8)	4.1(-12)	3.69(-7)
Probability (Non-Regional Disasters)	1.35(-8)	1.35(-12)	1.35(-8)
Total Probability	5.45(-8)	5.45(-12)	3.83(-7)

*9.87(-1) = 9.87×10^{-1}

Table 5.8 Summary of source terms for damage state I-T for input to LGS-FES

Failure Modes and Release Paths	I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE	I-T/HB	I-T/LGT	I-T/$\overline{LGT}$
Xe-Kr	1.0	1.0	1.0	1.0	1.0	0.73	0.73
OI	6.99(-3)*	6.99(-3)	6.99(-3)	-	-	-	-
I	1.78(-3)	1.48(-4)	2.09(-4)	9.6(-2)	2.0(-1)	2.7(-3)	1.9(-2)
Cs	1.88(-2)	3.11(-4)	9.19(-4)	1.0(-1)	6.0(-2)	9.8(-5)	9.8(-2)
Te	8.41(-2)	1.23(-3)	2.16(-3)	4.0(-1)	1.0(-1)	4.6(-4)	4.6(-2)
Ba	9.94(-4)	1.91(-5)	8.22(-5)	1.0(-2)	7.0(-3)	1.6(-5)	1.6(-3)
Ru	4.95(-3)	7.39(-5)	1.39(-4)	4.0(-1)	8.0(-2)	3.2(-5)	3.2(-3)
La	9.89(-4)	1.46(-5)	2.61(-5)	2.0(-3)	1.0(-5)	5.8(-6)	5.8(-4)
Time of Release (hr)	5.17	5.17	5.17	2.4	2.4	1.5	1.5
Duration of Release (hr)	0.5	0.5	0.5	0.5	0.5	3.4	3.4
Warning Time (hr)	3.67	3.67	3.67	1.0	1.0	0	0
Energy of Release (10^6 Btu/hr)	100	100	100	130	100	1.0	1.0
Height (ft)	82	82	82	82	82	82	82
Probability (Regional Disasters)	5.6(-7)	5.1(-7)	5.7(-8)	2.3(-10)	2.3(-8)	5.0(-7)	6.2(-7)
Probability (Non-Regional Disasters)	1.99(-5)	1.80(-5)	2.02(-6)	8.08(-9)	8.08(-7)	1.79(-5)	2.21(-5)
Total Probability	2.05(-5)	1.85(-5)	2.03(-6)	8.31(-8)	8.31(-7)	1.84(-5)	2.27(-5)

*6.99(-3) = 6.99×10^{-3}

Table 5.9 Summary of source terms for damage state II-T for input to LGS-FES

Failure Modes and Release Paths	II-T/WW	II-T/SE
Xe-Kr	9.8(-1)*	1.0
OI	6.86(-3)	-
I_2	6.73(-1)	9.6(-2)
Cs	3.36(-1)	1.0(-1)
Te	2.31(-1)	4.0(-1)
Ba	4.1(-2)	1.0(-2)
Ru	4.0(-2)	4.0(-1)
La	3.28(-3)	2.0(-3)
Time of Release (hr)	24.92	27
Duration of Release (hr)	3.91	0.5
Warning time (hr)	5.32	7
Energy of Release (10^6 Btu/hr)	1.0	130.0
Height (ft)	82	82
Probability (Regional Disasters)	2.0(-8)	2.0(-12)
Probability (Non-Regional Disasters)	1.91(-6)	1.9(-10)
Total Probability	1.93(-6)	1.9(-10)

*9.8(-1) = 9.8×10^{-1}

Table 5.10 Summary of source terms for damage state III-T for input to LGS-FES

Failure Modes and Release Paths	III-T/WW	III-T/SE	III-T/HB	III-T/LGT	III-T/$\overline{\text{LGT}}$
Xe-Kr	1.0	1.0	1.0	7.3(-1)	7.3(-1)
OI	6.99(-3)*	-	-	-	-
I_2	7.81(-2)	4.0(-1)	2.0(-1)	2.7(-3)	1.9(-2)
Cs	2.24(-1)	4.0(-1)	6.0(-2)	9.8(-5)	9.8(-2)
Te	5.74(-1)	5.0(-1)	1.0(-1)	4.6(-4)	4.6(-2)
Ba	1.95(-2)	5.0(-2)	7.0(-3)	1.6(-5)	1.6(-3)
Ru	3.65(-2)	5.0(-1)	8.0(-2)	3.2(-5)	3.2(-3)
La	6.92(-3)	3.0(-3)	1.0(-5)	5.8(-6)	5.8(-4)
Time of Release (hr)	2.67	2.0	2.0	0.5	0.5
Duration of Release (hr)	1.38	0.5	0.5	3.5	3.5
Warning Time (hr)	2.17	1.0	1.0	0	0
Energy of Release (10^6 Btu/hr)	100	130	100	1.0	1.0
Height (ft)	82	82	82	82	82
Probability (Regional Disasters)	3.7(-7)	7.4(-11)	7.4(-9)	1.6(-7)	2.0(-7)
Probability (Non-Regional Disasters)	1.58(-6)	3.24(-10)	3.24(-8)	7.14(-7)	8.76(-7)
Total Probability	1.95(-6)	3.98(-10)	3.98(-8)	8.74(-7)	1.08(-6)

*6.99(-3) = 6.99×10^{-3}

Table 5.11 Summary of source terms for damage state IV-T for input to LGS-FES

Failure Modes and Release Paths	IV-T/DW	IV-T/WW	IV-T/$\overline{WW}$	IV-T/SE
Xe-Kr	9.99(-1)	9.99(-1)	9.98(-1)	1.0
OI	6.95(-3)*	6.95(-3)	6.95(-3)	-
I_2	4.74(-1)	4.61(-1)	4.68(-1)	4.0(-1)
Cs	4.86(-1)	4.81(-1)	5.18(-1)	4.0(-1)
Te	5.09(-1)	4.45(-1)	4.81(-1)	5.0(-1)
Ba	5.54(-2)	5.60(-2)	5.96(-2)	5.0(-2)
Ru	8.85(-1)	7.81(-2)	8.31(-2)	5.0(-1)
La	6.82(-2)	6.03(-3)	6.51(-3)	3 0(-3)
Time of Release (hr)	1.13	1.13	1.13	2.0
Duration of Release (hr)	3.34	3.34	3.34	0.5
Warning Time (hr)	0.5	0.5	0.5	1.5
Energy of Release (10^6 Btu/hr)	1.0	1.0	1.0	130
Height (ft)	82	82	82	82
Probability (Regional Disasters)	4.7(-8)	4.27(-8)	4.75(-9)	9.5(-12)
Probability (Non-Regional Disasters)	1.63(-7)	1.46(-7)	1.63(-8)	3.25(-11)
Total Probability	2.1(-7)	1.89(-7)	2.1(-8)	4.2(-11)

*6.99(-3) = 6.99×10^{-3}

Table 5.12 Summary of source terms for damage states I-S and IV-A for input to LGS-FES

Failure Modes and Release Paths	I-S/DW	IV-A/DW
Xe-Kr	9.99(-1)*	9.96(-1)
OI	6.99(-3)	6.94(-3)
I_2	3.31(-3)	4.78(-1)
Cs	4.89(-3)	5.06(-1)
Te	2.80(-3)	5.18(-1)
Ba	6.01(-4)	5.76(-2)
Ru	2.87(-4)	8.86(-2)
La	4.01(-4)	6.95(-3)
Time of Release (hr)	5.11	1.17
Duration of Release (hr)	0.5	3.0
Warning Time (hr)	3.76	0.5
Energy of Release (10^6 Btu/hr)	100	1.0
Height (ft)	82	82
Probability (Regional Disasters)	-	-
Probability (Non-Regional Disasters)	3.76(-8)	5.0(-9)
Total Probability	3.76(-8)	5.0(-9)

*9.99(-1) = 9.99×10^{-1}

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BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: June 11, 1984

TO: W. T. Pratt

FROM: J. W. Yang JWY

SUBJECT: BNL-NUREG-33835

INTRODUCTION

An inconsistency has been found between the text and the numerical results in the subject BNL informal report. Specifically, the Class S accident sequence is described as being equivalent to a large break LOCA (refer to pages 3-6, 4-9, and 5-4) but the timing of events (Table 3.9) release fractions (Table 4.9) and source-terms (Table 5.7) are not consistent with a large break LOCA calculation. A number of calculations were performed for accident Class S to determine the sensitivity of the results to various primary system assumptions. The large break LOCA calculation resulted in slightly higher source terms and lower warning times, so that it was selected as the representative sequence for this class. The text reflects this decision; unfortunately, the numerical results do not. Tables 3.9, 4.9, and 5.7 have therefore been reproduced from BNL-NUREG-33835 and modified to reflect the large break LOCA assumptions.

The original source terms in BNL-NUREG-33835 were used by the NRC staff to perform a site consequence analysis in support of the Final Environmental Statement* related to the operation of the LGS. The modified Class S source terms attached to this memorandum should have been used rather than the original source terms in BNL-NUREG-33835. However, it can be demonstrated that if the modified source terms were used in place of the original source terms the overall risk, as calculated by the NRC staff in the LGS-FES, would not change significantly. The Class S sequences contribute to only 2% of the long-term damage indices (e.g., latent fatalities). Changes in the fission product release fractions indicated in Table 5.7 would not therefore significantly influence these damage indices at Limerick. In addition, differences in the warning time do not affect long-term damage indices. Class S sequences contribute to 20% of the early fatalities at Limerick. However, most of this contribution comes from seismically initiated events. The evacuation model used for regional disasters by the NRC staff assumes a 20-hour delay so that

*Final Environmental Statement related to the operation of Limerick Generating Station, Units 1 and 2, NUREG-0974.

differences in the warning times (which could influence early damage indices) are not important for these seismically initiated events. In addition, most of the contribution to early fatalities for Class S sequences is due to the S-$\overline{H20}$ sequences and differences between the modified and original release fractions are minimal for this sequence. In summary, the modified release fractions and warning times for the Class S sequences do not significantly change overall risk at the Limerick facility.

DISCUSSION OF RESULTS

For the S-H20/$\overline{WW}$ case (in which the water is assumed to drain from the vessel prior to core melt and the containment is assumed to fail in the wetwell below the water level) a comparison between the BNL releases and those reported in the LGS-SARA for the equivalent release (VR) shows reasonable agreement, except for OI, Ba, and La. These differences are all indications of differences in source term methodology. Fission product releases are based in part on NUREG-0772 in the LGS-SARA and on RSS methods in BNL-NUREG-33835. A decreased release of organic iodine and an increased release of the barium group are characteristic of NUREG-0772 releases relative to RSS releases.

However, for the S-H20/$\overline{WW}$ case (in which water is assumed to remain in the vessel during core degradation and the containment is assumed to fail in the wetwell below the water line) the BNL release fractions are lower than the equivalent LGS-SARA releases (VRH20). The largest discrepancy occurs for the OI, Te, Ba, and La groups. Discrepancies in the first three fission product groups are partially explained by differences in methodology.

The LGS-SARA release fractions for the case with water (VRH20) are predicted to be significantly higher than without water (VR). This is not the case for the equivalent BNL calculations. Only for the iodine group is there an increase of approximately a factor of four in the BNL calculations for the case with water in the bottom head compared to the case with no water. Furthermore (for the aerosol groups), it is seen that the species released during the melt release phase (Cs and Ba) are higher for the case with water in the bottom head (S-H20/$\overline{WW}$). However, species released primarily during core/concrete interactions (Te, Ru, and La) show an increased release for the case with no water present (S-$\overline{H20}$/WW). These effects are shown graphically on Figures 1 and 2. Figure 1 shows the variation with time of the I_2 release which shows a significant increase at the time of core slump. This same characteristic is also true of Cs and Ba. Figure 2 shows the variation with time of the Te release, which shows a rapid release during core/concrete interactions for the case with no water. The case with water shows an initial increase during core slumping followed by a reduced release rate during core/concrete interactions. The release characteristics for Ru and La are similar to the Te release.

JWY:jr/tr

Table 3.9 BNL analyses of Class S (T_SRPVRB)

Key Events (hours)	S-H2O	S-$\overline{H2O}$
Containment fails	0	0
Core melt begins	0.5	0.3
Core melt ends	1.2	1.3
Vessel head fails	2.5	1.3
70-cm penetration of floor	6.0	3.8

Table 4.9 Fission product release fractions for Class S

ASSESSMENT	LGS-SARA		MODIFIED CALCULATIONS	
FAILURE MODE	VRH20	VR	S-H20/$\overline{WW}$	S-$\overline{H20}$/$\overline{WW}$
OXIDATION RELEASE			Yes	Yes
Xe - Kr	1.0	1.0	9.99(-1)	9.99(-1)
Organic Iodine	3.0(-4)	3.0(-4)	6.99(-3)	6.99(-3)
I_2	5.0(-1)	1.0(-1)	2.2(-1)	5.4(-2)
Cs	7.3(-1)	3.3(-1)	3.7(-1)	3.2(-1)
Te	7.5(-1)	3.3(-1)	3.0(-1)	4.1(-1)
Ba	3.5(-1)	1.5(-1)	3.8(-2)	3.4(-2)
Ru	7.0(-2)	4.0(-2)	5.3(-2)	6.6(-2)
La	5.0(-2)	2.0(-2)	4.1(-3)	5.5(-3)
DF for I_2			1	1
DF for Aerosols			1	1
Core Melt Start	0.34	0.25	0.5	0.3
Core Melt End	0.34	0.25	1.2	1.3
1st Vap. Release	1.0	3.75	2.5	1.3
2nd Vap. Release			3.0	1.8
Vap. Release End			4.5	3.3
Containment Fail			0.0	0.0