LASALLE ILRT INTERVAL EXTENSION RISK ASSESSMENT

Prepared for:

Exelon.

Prepared by:

ERIN Engineering and Research, Inc.
an SKF Group Company

MARCH 2006

LASALLE COUNTY STATION

LASALLE ILRT

INTERVAL EXTENSION RISK ASSESSMENT

Prepared by: E T Burns Date: 3-24-06

Reviewed by: [illegible] Date: 3-27-06

Approved by: _____ Date: _____

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

FOREWORD

This analysis is to support a one time extension of the LaSalle Unit 2 ILRT interval from the currently approved 15 year interval to 15 years and 15 months in order to accommodate slight changes in the LaSalle refueling schedule.

A similar analysis was submitted for a one time extension for the Units 1 and 2 LaSalle ILRT interval in 2002. The one time extension for 15 years was approved by the NRC in 2003.

The current analysis is based on the 2003A PRA model which is the current LaSalle model of record and which is an update of the PRA that was used for the 2002 ILRT extension request.

The format of this risk assessment follows the template developed in the recent EPRI report, TR-1009325 Revision 1, dated December 2005. It, therefore, includes the explicit evaluation of the age effects on corrosion of the steel liner as part of the baseline risk metric calculation.

The enclosed analysis demonstrates the one time ILRT interval extension to 15 years and 15 months results in a very small change in risk.

Section 1

PURPOSE OF ANALYSIS

1.1 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to a one time extension of fifteen years plus fifteen months for LaSalle Unit 2.[1] The extension would allow for substantial cost savings as the ILRT could be deferred to a scheduled refueling outage for the LaSalle County Station.

The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [5].

The methodology and the format of this document follows the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the December 2005 EPRI Final Report. [17]

---

(1) Methodology development in EPRI TR-104285 [2] and updated in EPRI TR-1009325, Rev. 1 [17] is based on an ILRT interval extension from 10 years to 15 years. Using a value longer than 15 years slightly changes the quantitative results of the analysis.

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three tests in ten years to at least one test in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than normal containment leakage of 1La.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0 [1], and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals." [2]

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for LaSalle County Station.

The NEI Interim Guidance for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is updated in EPRI TR-1009325 Rev. 1 [17] which is the basis for this risk assessment. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment three times every ten years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.3 CRITERIA

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines "very small changes" in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 1.0E-6 per reactor year and increases in large early release frequency (LERF) less than 1.0E-7 per reactor year. Because the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1.0E-6 per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help demonstrate that key principles, such as the defense-in-

depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated.

In addition, the total annual risk (population dose in person-rem/yr) is examined to demonstrate the relative change in this parameter. (No criteria have been established for this parameter change.)

LaSalle does not credit containment overpressure for the mitigation of design basis accidents. The LaSalle BWR/5 ECCS pumps are designed to be able to pump saturated fluid.

Section 2

METHODOLOGY

2.1 RISK ASSESSMENT

Quantitative risk metrics are used to estimate the change in public risk associated with the proposed change to the Unit 2 ILRT interval. These quantitative risk metrics are calculated using the latest LaSalle Unit 2 Probabilistic Risk Assessment (PRA) models, i.e., the 2003A PRA model. The baseline results for the 2003A PRA model are as follows:

CDF = 6.64E-06/yr
LERF = 3.56E-07/yr

2.2 GENERAL STEPS

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years plus fifteen months. The approach is consistent with that presented in NEI Interim Guidance [3], EPRI TR-104285 [2], NUREG-1493 [6] and the Calvert Cliffs liner corrosion analysis [5]. The report generally follows the approach documented in the template from EPRI TR-1009325 (Rev. 1) dated December 2005 [17]. The analysis uses results from the latest LaSalle LERF analysis (from the 2003A PSA model) coupled with supplemental containment response analysis (see Appendix B) resulting in a spectrum of fission product release categories including intact containment states. This risk assessment is directly applicable to the LaSalle County Station.

The six (6) general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years plus fifteen months (16.25 years).
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [4] and compare with the acceptance guidelines of RG 1.174.
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Additionally, the following items are noted:

- Consistent with the other industry containment ILRT extension risk assessments, the LaSalle assessment uses population dose as one of the risk measures. The other risk measures used in the LaSalle assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for LaSalle uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the NEI Interim Guidance and the Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [17].
- LaSalle does not credit containment overpressure for the mitigation of design basis accidents. The LaSalle BWR/5 ECCS pumps are designed to be able to pump saturated fluid.

2.3 PRA QUALITY

2.3.1 PRA Model Summary

The 2003A PRA model is the most recent evaluation of the risk profile at LaSalle Generating Station (LS) for internal event challenges. The LaSalle PRA was originally submitted to the NRC in April 1994 as the LaSalle Individual Plant Examination (IPE)

Submittal in response to NRC Generic Letter 88-20. The basis for the LaSalle IPE submittal was the PRA performed for the LaSalle plant by Sandia National Laboratories and documented in NUREG/CR-4832, *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP).* The LS PRA has been updated multiple times since the original IPE. A summary of the LS PRA history is as follows:

PRA Model Version	Issue Date	Total CDF (/yr)
LS IPE Submittal	April 1994	4.74E-5
Revision 1996	1996	1.05E-5
Revision 1999	November 1999	8.58E-6(1)
Revision 2000A	January 2000	5.90E-6
Revision 2000B	March 2000	5.90E-6
Revision 2000C	June 2000	8.20E-6
Revision 2001A	August 2001	5.70E-6
Revision 2003A	June 2003(2)	6.64E-6

Notes:
(1) Does not include internal floods.
(2) An update to 2003A was performed in May 2004 to document use of a newer version of CAFTA, but the results are unchanged and model revision remains at 2003A.

The LaSalle PRA modeling is a detailed PRA, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the LaSalle PRA is based on the event tree / fault tree methodology, which is a standard methodology in the PRA industry.

The PRA model version used for the ILRT interval extension is the average maintenance at-power PRA model of record, LaSalle PRA Revision 2003A.

2.3.2 PRA Peer Review: Peer Certification A & B Facts & Observations (F&Os)

The LaSalle internal events PRA Peer Review was completed and documented in July 2000. The purpose of the PRA Peer Review process is to provide a method for establishing the technical quality of a PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The PRA Peer Review process uses a review team composed of PRA and system analysts, each with significant expertise in both PRA development and PRA applications. This team provides both an objective review of the PRA technical elements and a subjective assessment, based on their PRA experience, regarding the acceptability of the PRA elements. The team uses a set of checklists as a framework within which to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA products available.

The LaSalle PRA Peer Review resulted in zero (0) "A" priority F&Os and fifteen (15) "B" priority F&Os, plus other lower priority F&Os (i.e., "C" and "D" priority – minor technical and editorial comments).

Thirteen (13) of the fifteen (15) "B" F&Os have been resolved in the LaSalle PRA updates that have been performed since the 2000 Peer Review. Archival documentation of the resolution of these items is maintained and available for review. The two (2) open "B" priority F&Os are documentation issues that do not impact the ILRT risk input.

2.3.3 Assessment of ASME Standard Supporting Requirements

The ASME PRA Standard is now published along with the NRC endorsement in RG 1.200. As part of the PRA update process for LaSalle, the LaSalle PRA has been subjected to a self assessment, using the NEI 00-02 process as modified by Appendix B of RG 1.200 for the ASME PRA Standard Supporting Requirements listed in Table 4 of Appendix G of NEI 99-02. The self assessment provided a roadmap to those critical LaSalle PRA items that should be updated to Capability Category II for the 2003 PRA

update. Using the self-assessment, the LaSalle PRA model was updated to the 2003A model using the insights from the self-assessment. An additional self-assessment following the update has identified that the critical aspects of the LaSalle PRA 2003A model meet Capability Category II.

Section 3

GROUND RULES

The following ground rules are used in the analysis:

- The LaSalle Level 1 and LERF internal events PSA models provide representative results. The LaSalle Unit 2 PSA model is used explicitly in this risk assessment. The LaSalle PSA models include transients, LOCAs, internal flooding scenarios and seismic induced accident sequence.
- It is appropriate to use the LaSalle internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire events were to be included in the calculations. A brief LaSalle specific discussion of the effects of external hazards on the results is also provided.
- Dose results for the containment failures modeled in the PSA can be characterized by plant specific information provided in the 1992 NRC risk assessment of LaSalle, NUREG/CR-5305 [19]. The dose results for this analysis are estimated by scaling the NUREG/CR-5305 results by population differences for LaSalle since the 1992 study.
- The use of year 2000 population data is adequate for this analysis. Scaling the year 2000 population data to the estimated population of 2009 (the date of the ILRT extension) would not significantly impact the quantitative results, nor would it change the conclusions.
- Accident classes describing radionuclide release end states are defined consistent with the EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is $1L_a$ (L_a is the Technical Specification maximum allowable containment leakage rate of 0.635% volume/day).
- EPRI Class 3 accounts for increased leakage due to Type A inspection failures.
  - The representative containment leakage for Class 3a sequences is $10L_a$ based on the previously approved methodology performed for Indian Point Unit 3 [8, 9].

- The representative containment leakage for Class 3b sequences is $35L_a$ based on the previously approved methodology [8, 9].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [8, 9].
- Corrosion of the steel liner due to age related effects is also accounted for in the Class 3 frequency calculations.

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Because the containment bypass contribution to population dose is not influenced by the change in ILRT frequency, no changes in the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

Section 4

INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 GENERAL RESOURCES AVAILABLE

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [14]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. NUREG-1150 [15] and NUREG/CR-4551 [7]
9. NUREG-5305 [19]
10. NEI Interim Guidance [3]
11. Calvert Cliffs liner corrosion analysis [5]
12. EPRI TR-1009325 [17]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different

containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that can be used as the bases for the consequence analysis of the ILRT interval extension for plants that do not have plant-specific dose calculations. The ninth study provides an ex-plant consequence analysis for a 50-mile radius surrounding the LaSalle County Station conducted by Sandia National Laboratory and therefore, provides plant specific dose calculations for the ILRT interval extension. The tenth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The eleventh study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the twelfth study provides a risk impact assessment template for documenting an evaluation of extended integrated leak rate test intervals including the age related corrosion of steel liners.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on four (4) events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all four (4) of the identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as LaSalle. NUREG/CR-4220 identifies inerted BWRs as having significantly improved potential for leakage detection because of the requirement to remain inerted during power operation. This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

NUREG-1273 [12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [14]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from three (3) per 10 years to one (1) per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

NUREG-1493 [6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined

IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

EPRI Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "... the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year ..."

NUREG-1150 [15] and LaSalle RMIEP [19]

Two options exist for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [15, 7] dose calculations were used in the EPRI TR-104285 [2] and TR-1009325 [19] studies, as discussed previously. The use of generic dose information for NUREG-1150 makes the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PSA.

Although LaSalle does not maintain a Level 3 PSA, a plant-specific Level 3 PSA was performed for the LaSalle plant by Sandia National Laboratories in the 1990 time frame. This study is documented in NUREG/CR-5305. [19]

This NUREG/CR-5305 ex-plant consequence analysis is calculated for the 50-mile radial area surrounding LaSalle, and is reported in total person-rem for discrete accident categories (termed Accident Progression Bins (APB) in NUREG/CR-5305). The NUREG/CR-5305 consequences may be utilized in this ILRT risk assessment provided the following adjustments are performed:

- Adjust the person-rem results to account for changes in:
  - Population
  - Reactor Power Level
  - Technical Specification Allowed Containment Leakage Rate
- Assign the adjusted NUREG/CR-5305 APB consequences to the EPRI categories used in this risk assessment

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs PWR analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. LaSalle also has a concrete containment (Mark II) with a steel liner.

EPRI TR-1009325 [17]

The EPRI study TR-1009325 is the most recent EPRI study relating to risk impact assessment of extended leak rate testing intervals. The EPRI report provides a methodology and documentation template that may be utilized for such an assessment.

4.2 PLANT-SPECIFIC INPUTS

The plant-specific information used to perform the LaSalle ILRT Extension Risk Assessment includes the following:

- Unit 2 Level 1 Model results [18]
- Unit 2 LERF Model results [18] and supplemental release path analysis in Appendix B
- Release category definitions used in the Level 2 Model [18]
- LaSalle-specific ex-plant consequence from NUREG/CR-5305 (RMIEP) [19]
- LaSalle specific population within a 50-mile radius developed in Appendix A
- ILRT results which demonstrate adequacy of the administrative and hardware issues [24][(1)]

(1) The two most recent Type A tests at LaSalle Unit 1 and Unit 2 have been successful, so the current Type A test interval requirement is 10 years.

- Containment failure probability data [18]

Unit 2 Level 1 PRA Model

The Level 1 PRA model (Rev. 2003A) that is used for LaSalle is characteristic of the as-built plant. The Level 1 model is a linked fault tree model, and was quantified with the total Unit 2 Core Damage Frequency (CDF) = 6.64E-6/yr. [18] Table 4.2-1 summarizes the pertinent LaSalle results in terms of functional accident class contributors to CDF.

As noted in Table 4.2-1, the CDF used in this ILRT evaluation is based on the sum of PRAQuant sequence results. The separate calculation of accident classes in PRAQuant results in the retention of some non-minimal sequences. This leads to a total CDF of 7.07E-6/yr slightly higher than the single top results. The use of the PRAQuant result leads to a slight conservatism in the risk metrics.

Unit 2 LERF Model

The LERF Model (Rev. 2003A) that is used for LaSalle was developed to calculate the LERF contribution. Appendix B extends the LERF-only model to calculate a full spectrum of radionuclide releases. The Appendix B models are then used to develop the radionuclide release category frequencies.

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-5305 (RMIEP) and adjusting the results as required to represent LaSalle today.

Each of the release categories of Table 4.2-2 is associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-5305. The collapsed APBs are characterized by various attributes related to the accident progression. Unique combinations of the various attributes result in a set of eight (8) bins that are relevant to the analysis. The definitions of the eight (8) collapsed APBs are provided in

NUREG/CR-5305 and are reproduced in Table 4.2-2 for reference purposes. Table 4.2-3 summarizes the NUREG/CR-5305 calculated population dose for LaSalle associated with each APB.

The NUREG/CR-5305 consequences summarized in Table 4.2-3 must be adjusted for use in this current analysis to account for changes in the following parameters:

- Population
- Reactor Power Level
- Technical Specification Allowed Containment Leakage Rate

Population Adjustment

As presented in Appendix A, the 50-mile radius population used in the 1992 NUREG/CR-5305 consequence calculations is 1,131,512 persons, whereas the year 2000 population within the 50-mile radius of LaSalle is estimated at 1,553,566 persons. This increase in population results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 1,553,566/1,131,512 = 1.37.

Reactor Power Level Adjustment

The reactor power level used in the NUREG/CR-5305 consequence calculations is 3293 MWth, whereas the current LaSalle full power level is 3489 MWth. This increase in reactor power level results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 3489/3293 = 1.06.

Containment Leakage Rate Adjustment

The containment leakage rate used in the 1992 NUREG/CR-5305 consequence calculations for core damage accidents with the containment intact is 0.5% over 24 hours, whereas the LaSalle maximum allowable containment leakage per Technical Specifications is 0.635% per day. While use of a leakage rate below the maximum allowable may be reasonable, this analysis assumes that containment leakage is at the

maximum allowable Technical Specification value. As such, this difference in allowable containment leakage rate results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 0.635/0.5 = 1.27. This adjustment factor applies only to the "no containment failure" cases (i.e., APBs #5 and #8).

NUREG/CR-5305 [19] Adjusted Doses

Table 4.2-4 summarizes the NUREG/CR-5305 doses after adjustment for changes in population, reactor power level, and containment leakage rate.

Application of LaSalle PSA Model Results to NUREG/CR-5305 Level 3 Output

The results of the LaSalle PSA Level 2 model are not defined in the same terms as reported in NUREG/CR-5305. In order to use the Level 3 model presented in that document, it is necessary to match the EPRI defined release categories to the collapsed APBs from NUREG/CR-5305. The assignments are shown in Table 4.2-5, along with the corresponding EPRI/NEI classes.

Table 4.2-5 defines the EPRI accident classes used in the ILRT extension evaluation, which is consistent with the EPRI/NEI methodology [2, 17]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report. The NUREG/CR-5305 consequence bins are assigned based on conservative evaluations of the EPRI defined conditions.

Table 4.2-1

LASALLE 2003A PSA MODEL ACCIDENT CLASS FREQUENCIES [18]
AS INPUTS TO THE LEVEL 2 PRA

Release Category	Definition	CDF Frequency/yr
IA	Loss of Makeup at High RPV Pressure	2.42E-7
IBE	Early Station Blackout	5.67E-7
IBL	Late Station Blackout	7.13E-7
IC	Loss of Makeup accidents involving mitigated ATWS scenarios	6.41E-9
ID	Loss of Makeup at Low RPV Pressure (Transient Initiators)	1.42E-6
II	Loss of Decay Heat Removal	3.65E-6
IIIA	LOCA accidents in which there is a loss of the RPV	1.00E-9
IIIB	SLOCA or MLOCA accidents in which RPV pressure is high at the time of core damage	9.39E-9
IIIC	Loss of Makeup at Low RPV Pressure (large LOCA Initiators)	5.70E-8
IIID	Large LOCA accidents with failure of the vapor suppression function	7.29E-8
IV	ATWS	1.61E-7
V	Containment Bypass	1.71E-7
	Total	7.07E-6[1]

[1] The accident class subtotals are calculated by merging the cutsets from each accident sequence contributing to the accident class. Their total of 7.07E-6 which is used in this analysis is slightly higher than the official "single top" CDF model estimate of 6.64E-6.

The CDF used in the ILRT analysis is the result of the PRAQuant calculation used to develop the Accident Class results. This calculation has some non-minimal accident sequences and the resulting CDF is 7.07E-6/yr. This CDF is slightly conservative for the purpose of this analysis.

Table 4.2-2

SUMMARY ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [19]

Summary APB Number	Description
1	**VB, Early CF, RPV at Low Pressure:** Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at low pressure at the time of vessel breach.
2	**VB, Early CF, RPV at High Pressure:** Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at high pressure at the time of vessel breach.
3	**VB, Late CF:** Vessel breach occurs and the containment fails late in the accident (i.e., hours after vessel breach).
4	**VB, Early or Late Venting:** Vessel breach occurs and the containment is either vented before vessel breach or late in the accident.
5	**VB, No CF:** Vessel breach occurs; however, the containment neither fails nor is vented during the accident.
6	**No VB, CF:** The core damage process is arrested (i.e., no vessel breach); however, the containment still fails during the accident due to the generation of steam and non-condensibles during the accident.
7	**No VB, Venting:** The core damage process is arrested before vessel failure. However, the containment is vented either before the onset of core damage or during the core damage process.
8	**No VB, No CF, No Venting:** The core damage process is arrested and the containment remains intact.

Table 4.2-3

CALCULATION OF LASALLE POPULATION DOSE RISK AT 50 MILES [19][(1)]

Collapsed Bin #	Fractional APB Contributions to Risk[(3)]	NUREG/CR-5305 Population Dose Risk at 50 miles[(4)]	NUREG/CR-5305 Collapsed Bin Frequencies (per year)[(2)]	NUREG/CR-5305 Population Dose at 50 miles (person-rem)[(5)]
1	0.18	12.0	1.53E-05	7.85E+05
2	0.25	16.5	1.94E-05	8.51E+05
3	0.10	6.86	9.46E-06	7.26E+05
4	0.43	28.3	3.84E-05	7.37E+05
5	0.001	0.066	5.82E-06	1.13E+04
6	0.00	0	0.00E+00	N/A
7	0.03	1.91	9.05E-06	2.11E+05
8	0.001	0.066	6.76E-06	9.76E+03
Totals	1.00	66	1.04E-04	--

(1) This table is presented in the form of a calculation because NUREG/CR-5305 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose rate results.

(2) The total (i.e., internal plus external accident sequences) CDF of 1.04E-4/yr and the CDF subtotals by APB are taken from Figure 3.5-8 of NUREG/CR-5305.

(3) The individual APB contributions to total (i.e., internal plus external accident sequences) 50-mile radius dose rate are taken from Table 6.3-2 of NUREG/CR-5305.

(4) The individual APB 50-mile dose rates are calculated by multiplying the individual APB dose rate contributions by the total 50-mile radius dose rate of 66 person-rem/yr (taken from Table 6.2-1 of NUREG/CR-5305).

(5) The individual APB doses are calculated by dividing the individual APB dose rates by the APB frequencies.

(6) As the frequency of APB#6 was calculated as negligible (i.e., no frequency results survived the quantification truncation limit) in NUREG/CR-5305, no dose result can be estimated for APB#6.

Table 4.2-4

ADJUSTED NUREG/CR-5305 50-MILE RADIUS POPULATION DOSES

APB #	50-Mile Radius Dose (Person-rem) [5]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	Adjusted 50-Mile Radius Population Dose (Person-rem)
1	7.85E+05	1.37	1.06	n/a	1.14E+06
2	8.51E+05	1.37	1.06	n/a	1.24E+06
3	7.26E+05	1.37	1.06	n/a	1.05E+06
4	7.37E+05	1.37	1.06	n/a	1.07E+06
5	1.13E+04	1.37	1.06	1.27	2.09E+04
6	n/a	n/a	n/a	n/a	n/a
7	2.11E+05	1.37	1.06	n/a	3.07E+05
8	9.76E+03	1.37	1.06	1.27	1.80E+04

Table 4.2-5

EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

EPRI Class	Description	NUREG/CR-5305 APB
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant	#5[(1)]
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.	#2
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.	Est.[(2)] (#5)
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.	--[(4)]
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.	--[(4)]
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.	--[(4)]
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.	#4, 3, 3, 2, 1[(3)]
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.	#2

(1) APB #8 represents this class when no RPV breach occurs. It is conservatively subsumed by the use of APB #5.

(2) Uses multiples of APB #5 as directed by the EPRI/NEI methodology.

(3) The assignment of the Accident Progression Bin (APB) from NUREG/CR-5305 for the LaSalle severe accident spectrum is discussed in the text.

Class	APB
7a	#4
7b	#3
7c	#3
7d	#2
7e	#1

(4) Not affected by ILRT frequency and not analyzed in this assessment

LaSalle Population Dose By EPRI Category

The NUREG/CR-5305 dose results summarized in Table 4.2-4 are then assigned to the EPRI accident categories based on similarity of accident characteristics. The LaSalle 50-mile population dose by EPRI accident category are summarized in Table 4.2-6.

The dose for the "no containment failure" category (EPRI Category 1) is based on NUREG/CR-5305 APB #5. Two "no containment failure" APBs, one with RPV breach (APB #5) and one without RPV breach (APB #8), are analyzed in NUREG/CR-5305. The APB with the highest calculated 50-mile radius dose (i.e., the case with RPV breach, APB #5) is conservatively assigned to EPRI Category 1.

The dose for EPRI Category 2 is based on NUREG/CR-5305 APB #2. This assignment is based on assuming that the containment isolation failure of EPRI Category 2 occurs in the drywell. While APB #2 does not specify containment failure location, it results in the highest dose of all the NUREG/CR-5305 "containment failure" APBs (which is indicative of a drywell containment failure).

No assignment of NUREG/CR-5305 APBs is made for EPRI Categories 3a and 3b. Per the NEI Interim Guidance, the population doses for EPRI Categories #3a and #3b are taken as a factor of 10 and 35, respectively, times the population dose of EPRI Category 1.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of NUREG/CR-5305 APBs is made for these categories.

The dose for EPRI Category 7a is based on NUREG/CR-5305 APB #4. The majority of EPRI Category 7a is due to long-term loss of decay heat removal accidents in which

core damage, vessel breach, and containment failure in the wetwell airspace occur many hours after accident initiation.

The dose for EPRI Category 7b is based on NUREG/CR-5305 APB #3. The majority of EPRI Category 7b is due to loss of coolant make-up accidents in which core damage and vessel breach occur at low vessel pressure early in the accident, and containment failure in the drywell occurs many hours later.

The dose for EPRI Category 7c is also based on NUREG/CR-5305 APB #3. The majority of EPRI Category 7c is due to long-term loss of decay heat removal accidents in which core damage, vessel breach, and containment failure in the drywell occur many hours after accident initiation.

The dose for EPRI Category 7d is based on NUREG/CR-5305 APB #2. The LaSalle accident scenarios comprising EPRI Category 7d result in H/E release (the most severe release category). Accordingly, the most severe NUREG/CR-5305 dose case (i.e., APB #2) is used to characterize this category.

The dose for EPRI Category 7e is based on NUREG/CR-5305 APB #1. The majority of EPRI Category 7e is due to unmitigated ATWS accidents in which containment failure in the wetwell airspace, and subsequent core damage and vessel breach occur early in the accident scenario.

The dose for the containment bypass category, EPRI Category 8, is based on NUREG/CR-5305 APB #2. APB #2 results in the highest dose of all the NUREG/CR-5305 "containment failure" APBs, indicative (i.e., in a relative comparison to other accidents) of containment bypass scenarios.

The population dose rate evaluation using these population dose inputs is calculated in Section 5.

Table 4.2-6

LASALLE DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILE RADIUS[1]

EPRI Category	Category Description	NUREG/CR-5305 APB	Population Dose (Person-Rem Within 50 miles)
1	No Containment Failure	#5	2.09E+04
2	Containment Isolation System Failure	#2	1.24E+06
3a	Small Pre-Existing Failures	#5[2]	2.09E+05
3b	Large Pre-Existing Failures	#5[2]	7.32E+05
4	Type B Failures	--	n/a
5	Type C Failures	--	n/a
6	Other Containment Isolation System Failure	--	n/a
7a	Containment Failure Due to Severe Accident (a)	#4	1.07E+06
7b	Containment Failure Due to Severe Accident (b)	#3	1.05E+06
7c	Containment Failure Due to Severe Accident (c)	#3	1.05E+06
7d	Containment Failure Due to Severe Accident (d)	#2	1.24E+06
7e	Containment Failure Due to Severe Accident (e)	#1	1.14E+06
8	Containment Bypass Accidents	#2	1.24E+06

(1) The LaSalle estimates of population dose in person-rem are derived from the identified NUREG/CR-5305 Accident Progression Bin (APB) and includes the corrections for current power level, population update from NUREG/CR-5305, and current Tech. Spec. containment leakage as documented in Table 4.2-4.

(2) Multiplier of APB#5 are utilized as directed by the EPRI/NEI methodology to assess the population dose. (See text description and EPRI TR-1009325.)

4.3 IMPACT OF EXTENSION ON DETECTION OF COMPONENT FAILURES THAT LEAD TO LEAKAGE (SMALL AND LARGE)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class, as defined in Table 4.2-6, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the NEI Guidance [3]. For Class 3a, the probability is based on the mean failure from the available data (i.e., 5 "small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, a non-informative prior distribution is assumed for no "large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027). These probabilities are judged characteristic of the situation when ILRTs were performed at a frequency of 3 per 10 years.

In a follow on letter [20] to their ILRT guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and*

> *are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for LaSalle as detailed in Section 5, involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the LaSalle Level 2 PSA analysis.

Consistent with the NEI Guidance [3], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. Correspondingly, an extension of the ILRT interval to fifteen years and fifteen months can be estimated to lead to approximately a factor of 5.42 (8.125/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over estimates the factor increases attributable to the ILRT extension.

4.4 IMPACT OF EXTENSION ON DETECTION OF STEEL LINER CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The LaSalle primary containment is a pressure-suppression BWR/Mark II containment type that also includes a steel-lined reinforced concrete structure.

The liner sections at LaSalle are completely welded together and anchored into the concrete. There is no air space between the liner and the concrete structure. The corrosion/oxidation effects associated with water being in contact with the carbon steel liner and the concrete reinforcing bars are minimized due to the lack of available oxygen between the concrete and the liner. Furthermore, the liner is intended to be a membrane and constitutes a leak-proof boundary for the containment. The liner is nominally 0.25-inch thick and has been oversized to serve as form-work for concrete pouring during construction.

The following approach is used and documented in Table 4.4-1 to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment walls
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions: Base Model[1]

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4-1, Step 1.)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this LaSalle containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4-1, Step 1.)
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five (5) years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4-1, Steps 2 and 3.) Sensitivity studies in Section 6 are included that address doubling this rate every ten (10) years and every two (2) years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For LaSalle, the containment failure probabilities are conservatively assumed to be 1% for the drywell and wetwell vertical walls. Because the basemat for the LaSalle Mark II containment is in the suppression pool, it is judged that failure of this area would not lead to LERF. In any event, a 0.1% probability is assigned as a conservatism. Sensitivity studies in Section 6 are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4 for the Base Case values).

(1) Section 6 includes possible sensitivity cases for this effect.

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment walls. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. This is considered conservative since essentially 100% of the LaSalle containment interior surface is visible, whereas only 85% of the interior wall surface was estimated as being visible at Calvert Cliffs. Additionally, it should be noted that to date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1 , Step 5.) Sensitivity studies are included in Section 6 that evaluate total detection failure likelihoods as low as 5% and as high as 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder, truncated cone, and the containment basemat as summarized below.

Total Likelihood Of Non-Detected Containment Leakage Due To Corrosion:

- Case 1: At 3 years: 0.00071% + 0.00018% = 0.00089% = 8.90E-06
- Case 2: At 10 years: 0.00414% + 0.00103% = 0.00517% = 5.17E-05
- Case 3: At 16.25 years: 0.013% + 0.0033% = 0.0163% = 1.63E-04

The impact factor due to undetected corrosion is as follows for the three ILRT cases investigated:

Case 1: 3 ILRT per 10 years

$$\frac{\text{(3b Conditional Failure Probability} + \text{Total Likelihood of non-detected containment leakage due to corrosion at 3 yrs.)}}{\text{3b Conditional Failure Probability}} = \frac{2.70\text{E-3} + 8.9\text{E-6}}{2.7\text{E-3}} = 1.0033$$

Case 2: ILRT Per 10 years

$$\frac{9.0E\text{-}03 + 5.17E\text{-}05}{9.0E\text{-}3} = 1.00574$$

Case 3: ILRT per 16.25 years

$$\frac{1.4607E\text{-}2 + 1.63E\text{-}4}{1.4607E\text{-}2} = 1.0112$$

These impact factors are used to adjust the 3b accident class frequencies to model the impact of the undetected corrosion. See Table 5.1-1 under 3b for the example application for Case 1.

Analysis

Table 4.4-1

STEEL LINER CORROSION BASE CASE

Step	Description	Containment Cylinder, Truncated Cone, and Dome[1]		Containment Basemat	
1	**Historical Steel Liner Flaw Likelihood** Failure Data: Containment location specific (consistent with EPRI TR-1009325 and Calvert Cliffs analysis).	Events: 2 2/(70 tests * 5.5) = **5.2E-3** (Based on 20 units with lives over 5.5 years)		Events: 0 (assume 0.5 failure) 0.5/(70 tests * 5.5) = **1.3E-3** (Based on 20 units with lives over 5.5 years)	
2	**Age Adjusted Steel Liner/Shell Flaw Likelihood** During 15-year interval, assume failure rate doubles at the end of every five years (which equates to a 14.9% increase per year). The average over the 5th through 10th year period is set equal to the historical failure rate of Step 1 (consistent with Calvert Cliffs analysis). These assumptions are used to calculate the flaw likelihood for each year (for a 15 year plus 15 month period)	Year	Flaw Likelihood	Year	Flaw Likelihood
		0	1.79E-03	0	4.47E-04
		1	2.05E-03	1	5.13E-04
		2	2.36E-03	2	5.89E-04
		3	2.71E-03	3	6.77E-04
		4	3.11E-03	4	7.77E-04
		5	3.57E-03	5	8.93E-04
		6	4.10E-03	6	1.03E-03
		7	4.71E-03	7	1.18E-03
		8	5.41E-03	8	1.35E-03
		9	6.22E-03	9	1.55E-03
		10	7.14E-03	10	1.79E-03
		11	8.21E-03	11	2.05E-03
		12	9.43E-03	12	2.36E-03
		13	1.08E-02	13	2.71E-03
		14	1.24E-02	14	3.11E-03
		15	1.43E-02	15	3.57E-03
		15+15 mo.	1.70E-02	15+15 mo.	4.26E-03

Table 4.4-1

STEEL LINER CORROSION BASE CASE

Step	Description	Containment Cylinder, Truncated Cone, and Dome[1]	Containment Basemat
3	**Flaw Likelihood at 3, 10, and 15 years 15 months** This cumulative probability uses the age adjusted liner/shell flaw likelihood of Step 2 (consistent with Calvert Cliffs analysis – See Table 6 of Reference [5]). For example, the 7.12E-03 (at 3 years) cumulative flaw likelihood is the sum of the year 1, year 2, and year 3 likelihoods of Step 2.	**0.71% (1 to 3 years)** **4.14% (1 to 10 years)** **13.0% (1 to 15 years 15 months)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 years and 15 month intervals consistent with the desired presentation of the results.	**0.18% (1 to 3 years)** **1.03% (1 to 10 years)** **3.25% (1 to 15 years 15 months)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 years and 15 month intervals consistent with desired presentation of the results.
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.

Table 4.4-1

STEEL LINER CORROSION BASE CASE

Step	Description	Containment Cylinder, Truncated Cone, and Dome[1]	Containment Basemat
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1.0% * 10% **0.0041% (at 10 years)** 4.1% * 1.0% * 10% **0.013% (at 16.25 years)** 13.0% * 1.0% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0033% (at 16.25 years)** 3.3% * 0.1% * 100%

(1) The LaSalle containment dome is a steel drywell head that is 100% inspectable.

Section 5

RESULTS

The application of the approach based on NEI Interim Guidance [3], EPRI-TR-104285 [2], EPRI TR-1009325 [17], and previous risk assessment submittals on this subject [5, 8, 21, 22, 23] have led to a consistent analysis that produces LaSalle specific results. The LaSalle results are displayed according to the eight accident classes defined in the EPRI reports. Table 5-1 lists these EPRI accident classes.

The analysis performed examined LaSalle specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components, for example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1

EPRI ACCIDENT CLASSES [2, 3, 17]

EPRI Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight EPRI accident classes presented in Table 5-1.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight EPRI accident classes and provide the sum over all eight accident classes.

Step 3 - Evaluate risk impact in person-rem/yr of extending Type A test interval from approximately 3 to 16.25 years and 10 to 16.25 years.[1]

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

(1) Methodology development in EPRI TR-104285 [2] and updated in EPRI TR-1009325, Rev. 1 [17] is based on an interval extension from 10 years to 15 years. Using an interval longer than 15 years slightly changes the quantitative results of the analysis.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP)

5.1 STEP 1 - QUANTIFY THE BASELINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

The first step is to quantify the baseline frequencies for each of the EPRI accident categories. This portion of the analysis is performed using the LaSalle Level 1 and LERF PSA results supplemented by an expanded release pathway evaluation provided in Appendix B. The results for each EPRI category are described below.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). The frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as "OK" in the LaSalle radionuclide release end states of Appendix B).

The frequency of the LaSalle containment intact ("OK") accident bin is 1.02E-6/yr. As described below, the frequencies of EPRI Categories 3a and 3b are 6.85E-8/yr and 6.87E-9/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as 1.02E-6/yr – (6.85E-8/yr + 6.87E-9/yr) = 9.45E-7/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures; or random or common cause valve failures).

The frequency of this EPRI category is calculated as follows:

- Results (i.e., cutsets) of containment isolation failure fault tree (IS) are used as input
- All basic events, except those related to support system failure or random or common cause valve failures-to-close, are set to 0.00.
- Fraction of IS probability due to support system failure or random or common cause valve failures-to-close is then calculated. This value is then multiplied by the sum of the accident frequencies of the Level 2 containment isolation failure sequences (i.e., IA15, IBE15, IBL15, IC15, ID15, IE15, IIIA14, IIIB14, and IIIC14).

This process resulted in a fraction of 0.156 of the containment isolation failure probability due to support system failure or random or common cause valve failures-to-close. The sum of the LaSalle Level 2 containment isolation failure sequences is 1.78E-8/yr. Therefore, the frequency of EPRI Category 2 is 0.156 x 1.78E-8/yr = 2.78E-9/yr.

Note that all of the Level 2 containment isolation failure sequences outlined above except IBL15 are H/E sequences. Sequence IBL15 (representing 6.6E-10/yr of the EPRI Category #2 total frequency) is classified in the LaSalle Level 2 as a H/I release. This slight conservatism remains in the EPRI Category 2 calculation and is judged to have a negligible impact of the ILRT risk metrics associated with extending the test interval.

Class 3a and 3b are the methodology-directed categories that represent other isolation failures deemed affected by the IRLT frequency.

<u>Frequency of EPRI Category 3a</u>

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Consistent with NEI Interim Guidance [21], the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3, 17] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF. As discussed previously in Section 4.2, the LaSalle total core damage frequency is 7.07E-6/yr. Of this total CDF, the following core damage accidents involve either LERF directly (containment bypass) or will never result in LERF:

- Long-term Station Blackout (SBO) scenarios (LaSalle PSA Class IBL): 7.13E-7/yr[(1)]
- Loss of Containment Heat Removal accidents (LaSalle PSA Class II): 3.65E-6/yr[(2)]
- Containment Bypass accidents (LaSalle PSA Class V): 1.71E-7/yr[(3)]

Therefore, the frequency of EPRI Category 3a is calculated as (2.70E-02) x [(7.07E-6/yr) – (7.13E-7/yr + 3.65E-6/yr + 1.71E-7/yr)] = 6.85E-8/yr.

Frequency of EPRI Category 3b

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure or liner that would

(1) The long term SBO sequences lead to intermediate releases (Non-LERF).
(2) The loss of DHR sequences lead to intermediate releases (Non-LERF).
(3) The containment bypass sequences lead to early high releases (always LERF).

be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Similar to Category 3a, the frequency per year for this category is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3b failure probability (2.7E-3) value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Similar to EPRI Category 3a, the frequency of Category 3b is calculated as (2.70E-03) x [(7.07E-6/yr) – (7.13E-7/yr + 3.65E-6/yr + 1.71E-7/yr)] = 6.85E-9/yr.

This value is modified when considering the potential for undetected flaw growth. This is quantified in Section 4.4-1 for the base case.

The calculated impact of the undetected flaw in the steel liner for the base case of (3 ILRTs in 10 years) is the following as developed in Section 4.4:

$$\frac{\text{(3b Conditional Failure Probability} + \text{Total Likelihood of non-detected containment leakage due to corrosion at 3 yrs.)}}{\text{3b Conditional Failure Probability}} = \frac{2.70\text{E-3} + 8.9\text{E-6}}{2.7\text{E-3}} = 1.0033$$

The frequency of Category 3b considering the corrosion effects is calculated as (2.70E-03 x 1.0033) x [(7.07E-6/yr) – (7.13E-7/yr + 3.65E-6/yr + 1.71E-7/yr)] = 6.87E-9/yr.

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this

category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PSA results.

As the LaSalle LERF PSA enhanced for this analysis (refer to Appendix B) appropriately categorizes containment failure accident sequences into different release bins, EPRI Category 7 is sub-divided in this analysis to reflect the spectrum of the LaSalle Level 2 PSA results. The following sub-categories are defined here:

- Category 7a: severe accident induced containment failure resulting in Low magnitude releases in Intermediate time frame (LaSalle "L/I" release bin). The majority of EPRI Category 7a is due to long-term loss of decay heat removal accidents in which core damage, vessel breach, and containment failure in the wetwell airspace occur many hours after accident initiation.
- Category 7b: severe accident induced containment failure resulting in Moderate magnitude releases in Intermediate time frame (LaSalle "M/I" release bin). The majority of EPRI Category 7b is due to loss of coolant make-up accidents in which core damage and vessel breach occur at low vessel pressure early in the accident, and containment failure in the drywell occurs many hours later.
- Category 7c: severe accident induced containment failure resulting in High magnitude releases in Intermediate time frame (LaSalle "H/I" release bin). The majority of EPRI Category 7c is due to long-term loss of decay heat removal accidents in which core damage, vessel breach, and containment failure in the drywell occur many hours after accident initiation.
- Category 7d: severe accident induced containment failure resulting in High magnitude releases in Early time frame (LaSalle "H/E" release bin). The LaSalle accident scenarios comprising EPRI Category 7d result in H/E release (the most severe release category). Accordingly, the most severe NUREG/CR-5305 dose case (i.e., APB #2) is used to characterize this category.
- Category 7e: all other severe accident induced containment failure scenarios not represented by categories 7a through 7d. The majority of EPRI Category 7e is due to unmitigated ATWS accidents in which containment failure in the wetwell airspace, and subsequent core damage and vessel breach occur early in the accident scenario.

The frequency of Category 7a is the total frequency of the LaSalle Level 2 PSA "L/I" release bin. Based on the LaSalle Level 2 PSA results summarized in Table B-7, the frequency of L/I (Category 7a) is 2.40E-6/yr.

The frequency of Category 7b is the total frequency of the LaSalle Level 2 PSA "M/I" release bin. Based on the LaSalle Level 2 PSA results summarized in Table B-7, the frequency of Category 7b is 2.01E-6/yr.

The frequency of Category 7c is the total frequency of the LaSalle Level 2 PSA "H/I" release bin minus the portion of the EPRI Category 2 frequency resulting in H/I releases. Based on the LaSalle Level 2 PSA results summarized in Table B-7, the frequency of Category 7c is calculated as 9.03E-7/yr. (The frequency of EPRI Category 2 resulting in H/I releases is 6.6E-10/yr and is negligible when calculating the frequency of Category 7c.)

The frequency of Category 7d is determined by subtracting from the total frequency of the LaSalle Level 2 PSA "H/E" release bin the frequency of EPRI Category 8 and the portion of the EPRI Category 2 frequency resulting in H/E releases. Based on the LaSalle Level 2 results summarized in Table B-7, the frequency of the LaSalle Level 2 PSA "H/E" release bin is 3.55E-7/yr. As described previously, the frequency of EPRI Category 2 resulting in H/E releases is 2.78E-9/yr. As described below, the frequency of EPRI Category 8 is 1.71E-7/yr. Therefore, the frequency of Category 7d is calculated as (3.55E-7/yr) – (1.71E-7/yr + 2.78E-9/yr) = 1.81E-7/yr.

The frequency of Category 7e, 3.70E-7/yr, is determined by summing the frequencies of the remaining LaSalle Level 2 PSA release bins:

Release Category	Frequency (per yr)
LL/I:	2.68E-07
LL/L:	1.24E-08
M/E:	6.79E-08
M/L:	1.03E-08
LL/E:	1.09E-08
L/E:	0.00
L/L:	0.00
H/L:	0.00

The release characteristics of Category 7e is conservatively modeled by the Moderate/Early (M/E) LaSalle release bin.

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the LaSalle Level 1 PSA containment bypass scenarios (Class V). Based on the LaSalle Level 1 PSA results summarized earlier in Table 4.2-1, the frequency of Category 8 is 1.71E-7/yr.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The results are summarized in Table 5.1-1.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question regarding containment integrity is modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes are considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5.1-1 are developed for LaSalle by: (1) first determining the frequencies for Classes 2, 7, and 8 using the categorized EPRI classes and the identified correlations to APB shown in Table 4.2-5; (2) determining the frequencies for Classes 3a and 3b; and then, (3) determining the remaining frequency for Class 1. Furthermore, adjustments are made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the containment steel liner per the methodology described in Section 4.4.

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285, the NEI Interim Guidance and EPRI TR-1009325. Table 5.1-2 summarizes these accident frequencies by accident class for LaSalle Unit 2 both with and without the age related corrosion calculation for an ILRT frequency of 3 per 10 years.

Table 5.1-1

BASELINE[1] RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY (ILRT FREQUENCY OF 3 PER 10 YEARS)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated.	*[Total LaSalle "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b]* *[1.02E-6/yr] – [6.85E-8/yr + 6.87E-9yr] = 9.45E-7/yr*	9.45E-07
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	Cutsets of all LaSalle containment isolation fault tree used as input. All failure modes, except those related to support system failures or random and common cause valve failures-to-close, set to 0.00. Resulting fraction of IS failure probability due to support system or random or common cause FTC failures (0.156) multiplied by frequency sum of LaSalle CET containment isolation failure sequences (IA15, IBE15, IBL15, IC15, ID15, IE15, IIIA14, IIIB14, and IIIC14).	2.78E-09
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	*[LaSalle CDF for accidents not involving containment failure/bypass] x [2.7E-2]* *[(7.07E-6/yr) – (7.13E-7/yr + 3.65E-6/yr + 1.71E-7/yr)] x [2.70E-02] = 6.85E-8/yr*	6.85E-08
3b	Large Pre-Existing Failures[1]: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	*[LaSalle CDF for accidents not involving containment failure/bypass] x [2.7E-3 x Calculated impact of undetected flaw in steel liner]* *[(7.07E-6/yr) – (7.13E-7/yr + 3.65E-6/yr + 1.71E-7/yr)] x [2.70E-03 X 1.0033] = 6.87E-9/yr*	6.87E-09[1]

(1) Includes the age related corrosion effects on the steel liner.

Table 5.1-1

BASELINE[1] RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY (ILRT FREQUENCY OF 3 PER 10 YEARS)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from an ILRT (and thus not affected by ILRT testing frequency).	N/A (not affected by ILRT frequency)	n/a
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from an ILRT (and thus not affected by ILRT testing frequency).	N/A (not affected by ILRT frequency)	n/a
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities).	N/A (not affected by ILRT frequency)	n/a
7a	Containment Failure Due to Accident (a): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7a is defined in this analysis to apply to LaSalle PSA accidents that result in L/I releases. Not affected by ILRT leak testing frequency.	*[Total LaSalle "L/I" release category frequency]*	2.40E-06
7b	Containment Failure Due to Accident (b): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7b is defined in this analysis to apply to LaSalle PSA accidents that result in M/I releases. Not affected by ILRT leak testing frequency.	*[Total LaSalle "M/I" release category frequency]*	2.01E-06

Table 5.1-1

BASELINE[1] RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY (ILRT FREQUENCY OF 3 PER 10 YEARS)

<table>
<tr><th>EPRI Category</th><th>Category Description</th><th>Frequency Estimation Methodology</th><th>Frequency (1/yr)</th></tr>
<tr><td>7c</td><td><u>Containment Failure Due to Accident (c)</u>: EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7c is defined in this analysis to apply to LaSalle PSA accidents that result in H/I releases. Not affected by ILRT leak testing frequency.</td><td>[Total LaSalle "H/I" release category frequency]

[9.03E-7/yr]</td><td>9.03E-07</td></tr>
<tr><td>7d</td><td><u>Containment Failure Due to Accident (d)</u>: EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7d is defined in this analysis to apply to LaSalle PSA accidents that result in H/E releases (excluding contributions from EPRI Categories 2 and 8). Not affected by ILRT leak testing frequency.</td><td>[Total LaSalle "H/E" release category frequency] – [(Frequency EPRI Category #8)+(Portion of EPRI Category #2 frequency resulting in H/E)]

[3.55E-7/yr] – [1.71E-7/yr + 2.78E-9/yr] = 1.81E-7/yr</td><td>1.81E-07</td></tr>
<tr><td>7e</td><td><u>Containment Failure Due to Accident (e)</u>: EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7e is defined in this analysis to apply to LaSalle PSA accidents that result in all other remaining release categories (consequences modeled in this assessment by M/E releases). Not affected by ILRT leak testing frequency.</td><td>Calculated as the sum of all other remaining LaSalle release categories:
<table>
<tr><th>Category</th><th>Frequency (/yr)</th></tr>
<tr><td>LL/I:</td><td>2.68E-7</td></tr>
<tr><td>LL/L:</td><td>1.24E-8</td></tr>
<tr><td>M/E:</td><td>6.79E-8</td></tr>
<tr><td>M/L:</td><td>1.03E-8</td></tr>
<tr><td>LL/E:</td><td>1.09E-8</td></tr>
<tr><td>L/E:</td><td>0.00</td></tr>
<tr><td>L/L:</td><td>0.00</td></tr>
<tr><td>H/L:</td><td>0.00</td></tr>
</table></td><td>3.70E-07</td></tr>
<tr><td>8</td><td><u>Containment Bypass Accidents</u>: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.</td><td>[Total LaSalle Containment Bypass (Accident Class V) release frequency]</td><td>1.71E-07</td></tr>
<tr><td colspan="3">TOTAL:</td><td>7.07E-06</td></tr>
</table>

Table 5.1-2

RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (LASALLE BASE CASE) ILRT FREQUENCY OF 3/10 YEARS

EPRI Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	
		NEI Methodology	NEI Methodology Plus Corrosion[1]
1	No Containment Failure	9.45E-7	9.45E-07
2	Large Isolation Failures (Failure to Close)	2.78E-9	2.78E-09
3a	Small Isolation Failures (liner breach)	6.85E-8	6.85E-08
3b	Large Isolation Failures (liner breach)	6.85E-09	6.87E-09
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	See subclasses below:	See subclasses below:
7a	Containment Failure Due to Severe Accident (a)	2.40E-06	2.40E-06
7b	Containment Failure Due to Severe Accident (b)	2.01E-06	2.01E-06
7c	Containment Failure Due to Severe Accident (c)	9.03E-07	9.03E-07
7d	Containment Failure Due to Severe Accident (d)	1.81E-07	1.81E-07
7e	Containment Failure Due to Severe Accident (e)	3.70E-07	3.07E-07
8	Bypass (Interfacing System LOCA)	1.71E-07	1.71E-07
CDF	All CET end states	7.07E-06	7.07E-06

(1) Derived from Table 5.1-1

5.2 STEP 2 - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses are performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-5305 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3] are provided in Table 5.2-1.

Table 5.2-1

LASALLE POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES(1)

EPRI Category	Category Description	NUREG/CR-5305 APB	Population Dose (Person-Rem Within 50 miles)
1	No Containment Failure	#5	2.09E+04
2	Containment Isolation System Failure	#2	1.24E+06
3a	Small Pre-Existing Failures	#5(2)	2.09E+05
3b	Large Pre-Existing Failures	#5(2)	7.32E+05
4	Type B Failures	--	n/a
5	Type C Failures	--	n/a
6	Other Containment Isolation System Failure	--	n/a
7a	Containment Failure Due to Severe Accident (a)	#4	1.07E+06
7b	Containment Failure Due to Severe Accident (b)	#3	1.05E+06
7c	Containment Failure Due to Severe Accident (c)	#3	1.05E+06
7d	Containment Failure Due to Severe Accident (d)	#2	1.24E+06
7e	Containment Failure Due to Severe Accident (e)	#1	1.14E+06
8	Containment Bypass Accidents	#2	1.24E+06

(1) The LaSalle estimates of population dose in person-rem are derived from the identified NUREG/CR-5305 Accident Progression Bin (APB) and includes the corrections for current power level, population update from NUREG/CR-5305, and current Tech. Spec. containment leakage as documented in Table 4.2-4.

(2) Multiples of APB #5 are utilized as directed by the EPRI/NEI methodology

The above dose estimates, when combined with the results presented in Table 5.1-1, yield the LaSalle baseline mean consequence measure of dose rate (person-rem per reactor year) for each accident class. These results are presented in Table 5.2-2.

The baseline dose rates per EPRI accident category are calculated by multiplying the dose estimates summarized in Table 4.2-6 by the frequencies summarized in Table 5.1-1. The resulting baseline population dose rates by EPRI category are summarized in Table 5.2-1. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 5.2-1 are indicative of a 3-per-10 year ILRT surveillance frequency.

The LaSalle dose rate (person-rem/yr) compares favorably with other nuclear plant locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[9]
Peach Bottom	6.2	[21]
Farley Unit 2	2.4	[22]
Farley Unit 1	1.5	[22]
Crystal River	1.4	[23]
LaSalle	6.52	[Table 5.2-2]

Table 5.2-2

LASALLE ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

EPRI Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Dose Rate Change Due to Corrosion (Person-Rem/yr)[1]
			Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.09E+04	9.45E-07	0.02	9.45E-07	0.02	0
2	Large Isolation Failures (Failure to Close)	1.24E+06	2.78E-09	0.003	2.78E-09	0.003	0
3a	Small Isolation Failures (liner breach)	2.09E+05	6.85E-08	0.014	6.85E-08	0.014	0
3b	Large Isolation Failures (liner breach)	7.32E+05	6.85E-09	0.00501	6.87E-09	0.00503	2E-5
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Severe Accident Phenomena (Early and Late)		N/A	N/A	N/A	N/A	N/A
7a	Low/Intermediate Release	1.07E+06	2.40E-06	2.57	2.40E-06	2.57	0
7b	Moderate/Intermediate Release	1.05E+06	2.01E-06	2.11	2.01E-06	2.11	0
7c	High/Intermediate Release	1.05E+06	9.03E-07	0.95	9.03E-07	0.95	0
7d	High/Early Release	1.24E+06	1.81E-07	0.22	1.81E-07	0.22	0
7e	Moderate/Early Release	1.14E+06	3.70E-07	0.42	3.70E-07	0.42	0
8	Bypass (Interfacing System LOCA)	1.24E+06	1.71E-07	0.212	1.71E-07	0.212	0
CDF	All CET end states	--	7.07E-06	6.524[3]	7.07E-06	6.524[3]	2E-5

Notes to Table 5.2-2:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis based on the methodology of EPRI TR-1009325. [17]

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

(3) Round off in the calculation makes the sum slightly different than the printed values

5.3 STEP 3 - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS PLUS 15 MONTHS

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years and fifteen-months. To do this, an evaluation must first be made of the risk associated with the ten-year interval because the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Change Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). However, the frequency of Class 3a and 3b sequences is affected. The risk contribution from Classes 3a and 3b is increased for the ten year ILRT interval compared to the base case values by a factor of 3.33 based on the NEI guidance as described in Section 4.3. The results of the calculation for a 10-year interval are presented in Table 5.3-1. The change in population dose rate from the base case to the 10 year ILRT interval is 0.05 person-rem per year (i.e., 6.57 person-rem/yr – 6.52 person-rem/yr). This is judged to be a very small change.

Risk Change Due to 15 Year Plus 15 Month ILRT Interval

The risk metric changes for a 15 year plus 15 month ILRT interval compared with the base case is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the Class 3a and 3b frequencies factor of increase is 5.42 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-2. The change in population dose rate from the base case to the 15 year plus 15 month ILRT interval is 0.08 person-rem per year (i.e., 6.60 person-rem/yr – 6.52 person-rem/yr). This is judged to be a very small change.

Table 5.3-1

LASALLE ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

EPRI Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Dose Rate Change Due to Corrosion (Person-Rem/yr)[1]
			Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.09E+04	7.69E-07	0.0161	7.69E-07	0.0161	0
2	Large Isolation Failures (Failure to Close)	1.24E+06	2.78E-09	0.003	2.78E-09	0.003	0
3a	Small Isolation Failures (liner breach)	2.09E+05	2.28E-07	0.0477	2.28E-07	0.0477	0
3b	Large Isolation Failures (liner breach)	7.32E+05	2.28E-08	0.01669	2.293E-08	0.01679	9.54E-05
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Severe Accident Phenomena (Early and Late)		N/A	N/A	N/A	N/A	N/A
7a	Low/Intermediate Release	1.07E+06	2.4E-06	2.57	2.4E-06	2.57	0
7b	Moderate/Intermediate Release	1.05E+06	2.01E-06	2.11	2.01E-06	2.11	0
7c	High/Intermediate Release	1.05E+06	9.03E-07	0.95	9.03E-07	0.95	0
7d	High/Early Release	1.24E+06	1.81E-07	0.22	1.81E-07	0.22	0
7e	Moderate/Early Release	1.14E+06	3.7E-07	0.42	3.7E-07	0.42	0
8	Bypass (Interfacing System LOCA)	1.24E+06	1.71E-07	0.212	1.71E-07	0.212	0
CDF	All CET end states	--	7.07E-06	6.565[3]	7.07E-06	6.566[3]	9.54E-05

Notes to Table 5.3-1:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

(3) Round off in the calculation makes the sum slightly different than the printed values

Table 5.3-2

LASALLE ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/16.25 YEARS[3]

EPRI Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Dose Rate Change Due to Corrosion (Person-Rem/yr)[1]
			Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Class Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.09E+04	6.12E-07	0.013	6.11E-07	0.013	0
2	Large Isolation Failures (Failure to Close)	1.24E+06	2.78E-09	0.003	2.78E-09	0.003	0
3a	Small Isolation Failures (liner breach)	2.09E+05	3.71E-07	0.0775	3.71E-07	0.0775	0
3b	Large Isolation Failures (liner breach)	7.32E+05	3.71E-08	0.0272	3.7516E-08	0.02746	2.60E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal— Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Severe Accident Phenomena (Early and Late)		N/A	N/A	N/A	N/A	N/A
7a	Low/Intermediate Release	1.07E+06	2.4E-06	2.57	2.4E-06	2.57	0
7b	Moderate/Intermediate Release	1.05E+06	2.01E-06	2.11	2.01E-06	2.11	0
7c	High/Intermediate Release	1.05E+06	9.03E-07	0.95	9.03E-07	0.95	0
7d	High/Early Release	1.24E+06	1.81E-07	0.22	1.81E-07	0.22	0
7e	Moderate/Early Release	1.14E+06	3.7E-07	0.42	3.7E-07	0.42	0
8	Bypass (Interfacing System LOCA)	1.24E+06	1.71E-07	0.212	1.71E-07	0.212	0
CDF	All CET end states	--	7.07E-06	6.602[4]	7.07E-06	6.603[4]	2.60E-04

Notes to Table 5.3-2:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

(3) The one time ILRT extension request is for 15 years and a grace period of 15 months.

(4) Round off in the calculation makes the sum slightly different than the printed values

5.4 STEP 4 - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant risk metric is LERF.

For LaSalle, 100% of the frequency of Class 3b sequences is used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology).

5.4.1 LERF

The LERF calculations are performed including the penalty for corrosion. The LERF contribution calculated from EPRI Category 3b given the ILRT frequency of a 3/10 years is 6.87E-09/yr (Table 5.2-2); based on a ten-year test interval from Table 5.3-1, the Category 3b (LERF contribution) frequency is 2.293E-08/yr; and, based on a 16.25 year test interval from Table 5.3-2, the Category 3b frequency is 3.7516E-08/yr.

Therefore, the calculated increase in the LERF due to Class 3b sequences that is due to decreasing the ILRT test frequency from 3 in 10 years to 1 in 15 years plus 15 months (16.25 years) is 3.1E-08/yr (very small risk change). (Compare results from Table 5.3-2 with Table 5.2-2.)

Similarly, the increase in LERF due to decreasing the frequency from approximately 1 in 10 years to 1 in 15 years plus 15 months (16.25 years) is 1.46E-08/yr (very small risk change).

As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF for LaSalle is below the threshold criteria for a very small change when comparing the 15 year plus 15 month (16.25 years) results to either the current 10-year requirement, or to the original approximately 3-year requirement.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the NEI Interim Guidance and EPRI TR-1009325 [17]. The NRC has previously accepted similar calculations [9] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

$$CCFP = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency}) / CDF] * 100\%$$

$$CCFP_3 = 85.66\%$$

$$CCFP_{10} = 85.90\%$$

$$CCFP_{16.25} = 86.11\%$$

$$\Delta CCFP = CCFP_{16.25} - CCFP_3 = 0.45\%$$

$$\Delta CCFP = CCFP_{16.25} - CCFP_{10} = 0.21\%$$

The change in CCFP of 0.45% by extending the test interval to 15 years plus 15 months (16.25 years) from the original 3-in-10 year ILRT frequency requirement is judged to be an insignificant change in this risk metric.

5.6 SUMMARY OF RESULTS

The results from this ILRT extension risk assessment for LaSalle are summarized in Table 5.6-1.

Therefore, extending the ILRT interval from approximately three years to 16.25 years results in the following risk metric changes:

- ΔLERF = 3.1E-8/yr[1] (contribution to the change in LERF due to the corrosion effects is 4.0E-10/yr)
- ΔDose Rate = 0.08 person-rem/yr[2] (contribution to the change in dose rate due to the corrosion effects is 2.4E-04 person-rem/yr)
- CCFP = 0.45%

These changes in risk metrics are judged to be very small.

(1) The change in LERF from the base case to the 16.25 year ILRT interval is 3.1E-8/yr (i.e., 3.7516E-08 – 6.87E-09).

(2) The change in dose rate from the base case to the 15 year plus 15 month ILRT interval is 0.08 person-rem/yr (i.e., 6.60 person-rem/yr – 6.52 person-rem/yr).

Table 5.6-1

LASALLE ILRT CASES: (A) BASE (3.33 YR TEST INTERVAL);
(B) 10 YEAR TEST INTERVAL; (C) 16.25 YEAR TEST INTERVAL
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI Class	DOSE (Person-rem for 50-mile)	(A) Base Case ILRT Interval Approximately 3 Years		(B) ILRT Interval Extend to 10 Years		(C) ILRT Interval Extend to 16.25 Years	
		CDF (/Yr)	Dose Rate (Person-Rem/Yr)	CDF (/Yr)	Dose Rate (Person-Rem/Yr)	CDF (/Yr)	Dose Rate (Person-Rem/Yr)
1	2.09E+04	9.45E-07	0.0198	7.69E-07	0.0161	6.11E-07	0.013
2	1.24E+06	2.78E-09	0.003	2.78E-09	0.003	2.78E-09	0.003
3a	2.09E+05	6.85E-08	0.014	2.28E-07	0.0477	3.71E-07	0.0775
3b	7.32E+05	6.87E-09	0.00503	2.293E-08	0.01679	3.7516E-08	0.0275
7	1.07E+05	5.86E-06	6.27	5.86E-06	6.27	5.86E-06	6.27
8	1.24E+05	1.71E-07	0.212	1.71E-07	0.212	1.71E-07	0.212
Total	--	7.07E-06	6.524[1]	7.07E-06	6.566[1]	7.07E-06	6.603[1]
ILRT Dose Rate from Classes 3a and 3b (Person-rem/yr)		0.019		0.0645		0.105	
Delta Total Dose Rate (Person-Rem/yr)	From 3 yr	0		0.04		0.08	
	From 10 yr	--		0		0.040	
Class 3b Frequency[2] (LERF) (Per yr)		6.87E-09		2.293E-08		3.7516E-08	
Delta LERF (Per yr)	From 3 yr	0		1.61E-08		3.1E-08	
	From 10 yr	--		0		1.46E-08	
CCFP %		85.66%		85.90%		86.11%	
Delta CCFP %	From 3 yr	0		0.24%		0.45%	
	From 10 yr	--		0		0.21%	

(1) Round off in the calculation makes the sum slightly different than the printed values.
(2) Only the LERF associated with EPRI Class 3b is changing as a function of the ILRT interval. Therefore, that is all that is reported. The total LERF is 3.56E-07/yr.

Section 6

SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2, 5.3-1 and 5.3-2 show that including corrosion effects calculated using the assumptions described in Section 4.4 do not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases are developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. These sensitivities include the following:

- The time for the flaw likelihood to double is adjusted from every five years to every two and every ten years.
- The failure probabilities for the cylinder and dome and the basemat are increased and decreased by an order of magnitude.
- The total detection failure likelihood is adjusted from 10% to 15% and 5%.

For the corrosion sensitivity evaluation, a spread sheet is used to perform the calculation. This spreadsheet has used the total Class 7 frequency for LaSalle and the weighted average person-rem to perform the calculation. Because these values do not change in the sensitivity cases, no impact on the corrosion effects is to be seen.

Class 7	Dose (Person-Rem)	Frequency (/yr)	Dose (Person-Rem/yr)
7a Low/Intermediate Release	1.07E+06	2.4E-06	2.57
7b Moderate/Intermediate Release	1.05E+06	2.01E-06	2.11
7c High/Intermediate Release	1.05E+06	9.03E-07	0.95
7d High/Early Release	1.24E+06	1.81E-07	0.22
7e Moderate/Early Release	1.14E+06	3.7E-07	0.42
Total		5.86E-06	6.27

The values used for the Class 7 calculation are the total frequency for Class 7 and the weighted person-rem value to yield a value of 6.27 person-rem/yr.

Person-rem weighted average = 1.07E+06 person-rem

This corresponds to the total Class 7 frequency of 5.86E-6/yr.

The sensitivity case results are summarized in Table 6.1-1. In every case the impact from including the corrosion effects is minimal. Even the upper bound estimates with conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.8E-08/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis in Section 5 would not change.

6.2 ADDITIONAL SENSTIVITY CASES

6.2.1 Sensitivity To Class 3b Contribution To LERF

The Class 3b frequency for the base case of an equivalent three in ten-year ILRT frequency is 6.87E-09/yr [Table 5.2-2]. Extending the ILRT interval to ten years results in a frequency of 2.293E-08/yr [Table 5.3-1]. Extending the ILRT interval to 16.25 years results in a frequency of 3.7516E-08/yr [Table 5.3-2], which is an increase of 1.46E-08/yr relative to the 10 year-ILRT interval. If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF and the ILRT frequency is increased from an equivalent three in ten years ILRT to one in 16.25 year ILRT, then the increase in LERF is 3.1E-08/yr which is below the RG 1.174 threshold for very small changes in LERF of 1E-7/yr.

Table 6.1-1

STEEL LINER CORROSION SENSITIVITY CASES

Sensitivity Variables			Increase in Class 3b Frequency (LERF) for ILRT Interval Extension From Approximately 3 to 16.25 years (per Rx-yr)	
Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Cylinder, 0.1% Basemat)	Base Case 10%	3.1E-08	4.2E-10
Doubles every 2 yrs	Base	Base	3.15E-08	1.3E-09
Doubles every 10 yrs	Base	Base	3.1E-08	3.2E-10
Base	Base	*15%*	3.1E-08	5.9E-10
Base	Base	*5%*	3.1E-08	2.5E-10
Base	10% Cylinder, 1% Basemat	Base	3.4E-08	4.2E-09
Base	0.1% Cylinder, 0.01% Basemat	Base	3.0E-08	4.2E-11
Lower Bound				
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 1%	3.0E-08	1.3E-11
Upper Bound				
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%	4.8E-08	1.8E-08

6.2.2 Sensitivity to Treatment of Long Term SBO Sequences

The LaSalle long-term station blackout core damage accidents (Class IBL) result in non-LERF releases based on radionuclide release timing (i.e., LaSalle IBL core damage accidents have the potential to result in the entire spectrum of release magnitudes, including High magnitude releases; but, they can not result in Early releases). The following discussion focuses on the timing of long term station blackout (Class IBL) accident scenarios.

Typical of many industry PRAs, the LaSalle PRA uses a radionuclide release categorization scheme comprised of two factors: release timing and release magnitude. Three timing categories are used, as follows:

1. Early (E) — Less than 6 hours
2. Intermediate (I) — Greater than or equal to 6 hours, but less than 24 hours
3. Late (L) — Greater than or equal to 24 hours.

The definition of the categories is based upon past experience concerning offsite accident response:

- 0-6 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.
- 6-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- >24 hours are times at which the offsite measures can be assumed to be effective.

The timing categories are relative to the declaration of the LaSalle General Emergency Action Level (LaSalle procedure LZP-1200-1).

The LaSalle IBL accident scenarios include only those sequences in which high pressure injection (RCIC) is available initially in the accident but subsequently fails. The representative IBL sequence for LaSalle is sequence LOOP-17 of the LOOP event tree. Sequence LOOP-17 proceeds as follows (LaSalle MAAP run #LS008):

Event	Time After Plant Trip
- Loss of Offsite Power initiating event	0
- Failure of emergency AC power (EDGs)	0
- Failure of HPCS	0
- RCIC Initiation	~1 min.
- RPV/containment parameters exceed HCTL curve	7 hrs.
- Battery depletion	7 hrs.
- Failure to blowdown (no DC power)	7 hrs.
- Loss of RCIC (all) injection	7 hrs.
- Time for RPV level to drop to TAF	8.8 hrs.
- Time to core damage (1800F)	9.9 hrs.
- Time to energetic containment failure (fastest, but low frequency, release scenario)	~10 hrs.

As can be seen from the above scenario, the LaSalle IBL accident class results in a radionuclide release no earlier than 10 hours after the LOOP initiator. The 10 hour release for the IBL core damage accident makes the conservative assumption that an early energetic containment failure mode (in-vessel corium-steam explosion) occurs at about the time of core melt and relocation to the lower head (a low probability containment failure mode for the IBL accident).

LaSalle procedure EP-AA-1005 (Recognition Category MG1) directs declaration of a General Emergency (i.e., the emergency classification with associated directives for evacuation) for the following station blackout conditions:

- Loss of power from TR-241 and TR-242, and
- Emergency diesel generators fail to supply power to buses 241Y and 242Y, and
- Restoration of power to bus 241Y or 242Y within 4 hours is judged NOT likely.

The loss of offsite and emergency power to buses 241Y and 242Y occurs at t=0 for sequence LOOP-17. The LaSalle PRA assumes that the determination that AC power is not likely to be restored in the 4 hour time frame is made at approximately 1 hour into the accident. As such, a General Emergency is declared at 1 hour into the event. The evacuation process would be initiated within minutes after the declaration (i.e., LaSalle procedure EP-AA-111 states that local authorities must be notified within 15 minutes after the General Emergency declaration), and is likely to be completed within 4 hours based on site specific evacuation studies for weather and times of day variations. The earliest possible release for the IBL scenario occurs at approximately 10 hours (approximately 5 hours after evacuation is expected to be completed). Therefore, the IBL core damage accident is not an Early release.

Including long-term SBO scenarios in the EPRI Category 3a and 3b frequency calculations would not be typical or consistent with the NEI ILRT risk assessment methodology, but is performed here as a sensitivity study based on questions raised in previous RAIs. The results for the sensitivity case are discussed as follows:

- The calculated increase in LERF associated with a change in the ILRT frequency from the 3-in-10 year ILRT frequency to the 1-in-16.25 year frequency is determined to be 3.94E-08/yr, which remains below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk change.
- The calculated increase in population dose rate associated with a change in the ILRT frequency from the 3-in-10 year ILRT frequency to the 1-in-16.25 year frequency is determined to be 1.0E-01 person-rem/yr, which is an increase of 1.5% above the 3-in-10 year value of 6.53 person-rem/yr.
- The increase in the containment failure probability (CCFP) is determined to be 0.56%: (86.25% for the 1-in-16.25 year case versus 85.69% for the 1-in-3 year case).
- Note that these sensitivity case values include the concealed containment flaw evaluation.

Whether or not long term SBO scenarios are included in the EPRI Category 3a and 3b frequencies, the conclusion of the risk assessment does not change; that is, the LaSalle ILRT interval extension to 1-in-16.25 yr. has a minimal impact on plant risk.

6.3 POTENTIAL IMPACT FROM EXTERNAL EVENTS CONTRIBUTION

The impact of external events on this ILRT risk assessment is summarized in this section (refer to Appendix C for further detail). The following categories of external events are discussed:

- Seismic
- Internal Fires
- High winds/tornadoes
- External Floods
- Other

6.3.1 Overview of LaSalle External Events

Seismic Events

Seismic-induced accident sequences are included in the LaSalle Revision 2003A PSA; as such, they are included explicitly in the quantification of this ILRT risk assessment.

Internal Fires

LaSalle does not currently maintain PSA models for internal fires. The impact of internal fires on this ILRT risk assessment is based on review of the internal fires PSA work performed for LaSalle as part of the RMIEP study (NUREG/CR-4832). Refer to Appendix C.2 for a detailed discussion.

The LaSalle fire risk, as evaluated in the RMIEP study, is dominated by long term core damage accidents. The risk impact (LERF) of ILRT frequency changes is dominated by short term core damage accidents. As such, explicit inclusion of internal fire accident

frequency information in this ILRT risk assessment would not significantly alter the LERF quantitative results nor would it change the conclusions of this assessment.

High Winds/Tornadoes

The LaSalle plant design with respect to high wind and tornado loadings meets all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by high winds or tornadoes are not significant contributors to plant risk (approximately 1% of the Revision 2003A PSA CDF).

External Floods

The LaSalle plant design with respect to external flooding meets all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by external flooding are negligible contributors to plant risk.

Other External Hazards

The LaSalle site characteristics and design meet all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by transportation accidents, nearby facility accidents, turbine missiles, and other miscellaneous external hazards are not significant contributors to plant risk.

6.3.2 Qualitative Assessment of Impact on External Event Risk

Given the characteristics of this specific proposed plant change (i.e., ILRT interval extension), specific quantitative information regarding the impact on external event hazard risk measures is not a significant decision making input. The proposed ILRT interval extension impacts plant risk in a very specific and limited way, that is, it impacts a subset of accident sequences in which the probability of a pre-existing containment leak is the initial containment failure mode given a core damage accident. This impact is manifested in the plant risk profile in a similar manner for internal events and external events.

Although it is not possible at this time to incorporate quantitative risk assessments of all[1] external event hazards into this assessment, it is judged that if all external hazards were modeled in detail and a quantitative evaluation were performed in support of this proposed plant change, the calculated risk increase for both internal and external hazards would remain "very small".

[1] As discussed earlier, seismic-induced accident sequences are included explicitly in the quantitative analyses of this risk assessment.

Section 7

CONCLUSIONS

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test interval to fifteen years plus 15 months grace period (16.25 years):

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Because the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from three in ten years to one in 16.25 years is conservatively estimated as 3.1E-08/yr using the NEI guidance. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.
- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. Because the increase in LERF is significantly less than 1E-06/yr, this additional step is not required. Nevertheless, the total LERF is much less than 1E-5/yr.
- The change in population dose rate (person-rem/yr) associated with the change in Type A test frequency from 3 to 10 years to once-per-16.25-years is 0.08 person-rem/yr. Therefore, the risk impact when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure frequency from the three in ten year frequency to one in 16.25 year frequency is 0.45%. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

Therefore, increasing the ILRT interval to 16.25 years is considered to have an insignificant effect on the LaSalle metrics because it represents a very small change to the LaSalle risk profile.

Previous Assessments

The NRC in NUREG-1493 [4] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for LaSalle confirm these general findings on a plant specific basis considering the severe accidents evaluated for LaSalle, the Mark II containment failure modes, and the local population surrounding LaSalle County Station.

Section 8

REFERENCES

[1] Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, July 1995.

[2] Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

[3] Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

[4] An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, July 1998.

[5] Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.

[6] Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.

[7] Evaluation of Severe Accident Risks: Surry Unit 1, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.

[8] Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.

[9] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

[10] Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

[11] Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

[12] Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.

[13] Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

[14] Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

[15] Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG -1150, December 1990.

[16] United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

[17] K. Canavan, et. al., *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, EPRI TR-1009325 Revision 1, December 2005.

[18] LaSalle PRA Summary document, Revision 5, December 2003.

[19] Sandia National Laboratories, Integrated Risk Assessment for the LaSalle Unit 2 Power Plant; Phenomenology and Risk Uncertainty Evaluation Program (PRUEP), NUREG/CR-5305, June 1992 – May 1993 (4-volume set).

[20] Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.

[21] Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

[22] Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.

[23] Letter from D.E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

[24] NRC letter to LaSalle County Station Issuing Technical Specification Amendment No. 110 and 95, dated January 11, 1996 to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

Appendix A
LASALLE POPULATION DATA

The 50-mile radius population dose (person-rem) estimates used in this ILRT risk assessment are based on the LaSalle-specific accident consequence calculations documented in the 1992 NUREG/CR-5305 study. In order to use these 1992 LaSalle consequence results, they must first be scaled upward to account for the growth in population around the LaSalle site in the past decade.

A.1 NUREG/CR-5305 POPULATION

While the 1992 LaSalle NUREG/CR-5305 study reports population dose rate results for the 50-mile radius around the LaSalle site, the NUREG/CR-5305 documentation does not report the population total of the 50-mile radius used in the analysis. The purpose of this appendix is to estimate the 50-mile radius population total that was used in the NUREG/CR-5305 study, so that it may be used in this ILRT risk assessment for scaling and estimating population dose rates.

Table A-1 summarizes the population data around the LaSalle site as reported in the NUREG/CR-5305 study. As can be seen from Table A-1, this population data is for various radial distances around the plant, and does not include explicit information for the 50-mile radius.

Three methods are used here to estimate the 50-mile radius population used in the NUREG/CR-5305 study:

> Method 1: Using the NUREG/CR-5305 reported population data points, assume direct proportion of population with area
>
> Method 2: Using the NUREG/CR-5305 reported population data points, interpolate between estimates for 30 miles and 100 miles as a function of area

Table A-1

LASALLE POPULATION DATA REPORTED IN NUREG/CR-5305 [19]

Radius From Site		Population (persons) (1)
Miles	Kilometers	
1	1.6	24
3	4.8	309
10	16.1	14, 730
30	48.3	217, 620
100	160.9	10, 372, 934
350	563.3	48, 584, 604
1000	1609.3	179, 831, 712

(1) The NUREG/CR-5305 population estimates are based on 1980 census information, updated to reflect the time period of the NUREG/CR-5305 study.

Method 3: Using U.S. Census 2000 data and associated percentage changes in municipality populations compared to 1990 Census data, calculate the 1990 50-mile radius population

Method 1

This method assumes a constant population density, thus calculating the population of one area as a direct proportion of another. This population estimation method is performed for both the NUREG/CR-5305 30-mile radius data point and the 100-mile radius data point.

Using the population density indicated by the 30-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{30}^2}{217{,}620} = \frac{\pi R_{50}^2}{Pop_{50}}$$

$$Pop_{50} = 217{,}620 \times (R_{50}^2/R_{30}^2) = 604{,}500 \text{ persons}$$

Using the population density indicated by the 100-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{50}^2}{Pop_{50}} = \frac{\pi R_{100}^2}{10{,}372{,}934}$$

$$Pop_{50} = 10{,}372{,}934 \times (R_{50}^2/R_{100}^2) = 2{,}593{,}233 \text{ persons}$$

Using the 30-mile radius data point to calculate the 50-mile radius population produces a lower end value, as the population density closer to the site is comparatively low. Using the 100-mile radius data point produces a higher end value, as the population density for the 100-mile radius includes the highly populated Chicago area. The more correct value lies between these estimates.

Method 2

This population estimation method is an interpolation assuming a linearly increasing population with distance (refer to Figure A-1). Interpolating, using areas corresponding to the distances, results in the following 50-mile radius estimate;

$$\frac{(10{,}372{,}934 - 217{,}620)}{(3.14E{+}4 - 2.83E{+}3)} = \frac{(Pop_{50} - 217{,}620)}{(7.85E{+}3 - 2.83E{+}3)}$$

$$Pop_{50} = 2{,}001{,}998 \text{ persons}$$

Method 3

This population estimation method makes use of the 2000 U.S. Census information to back calculate the 50-mile radius population around the LaSalle site in the 1990 time frame. As discussed in the next section, the 2000 U.S. Census information has been analyzed in support of this study to estimate the 50-mile radius population for 2000. From that analysis the following information is available:

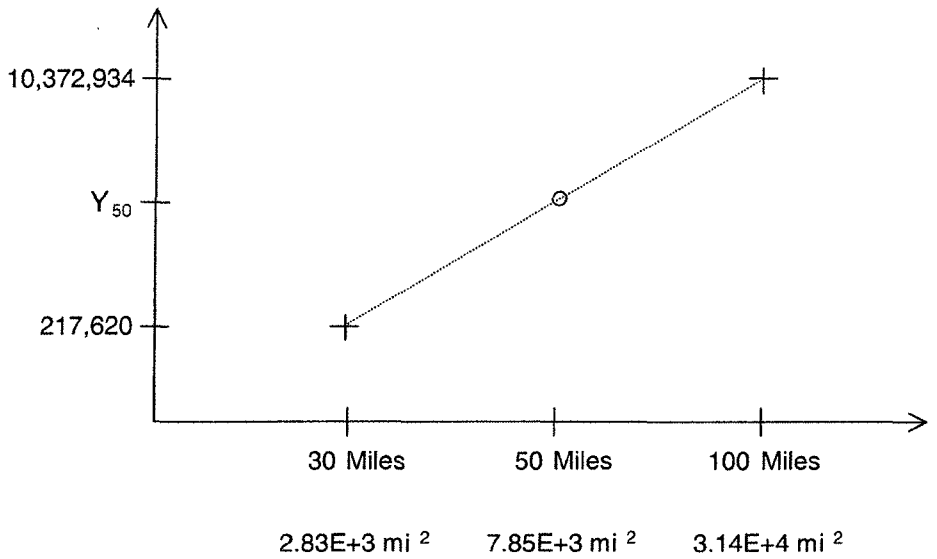
- 50-mile radius population around LaSalle for 2000
- Population change compared to 1990

As described in the following section, the 50-mile radius population around LaSalle for 2000 is estimated at 1,553,566 persons.

The 2000 U.S. Census data also provides population changes (compared to 1990 U.S. Census data) for discrete municipalities. Table A-2 provides a summary of discrete municipalities within the 50-mile radius of the LaSalle plant along with the population changes between 1990 and 2000. Table A-2 contains the majority of the city population within the 50 mile radius from LaSalle. The population of these discrete municipalities represents approximately 50-55% of the total population within the 50-mile radius of LaSalle. The total percentage change in population of the municipalities

Figure A-1

LINEAR RELATIONSHIP USED IN
NUREG/CR-5305 POPULATION ESTIMATION METHOD #2



in Table A-2 is assumed here to apply uniformly across the entire 50-mile radius. The assumption is made that the growth rate of these municipalities can be taken to be the growth rate for the entire population within 50 miles of LaSalle.

As can be seen from Table A-2, the percentage population change from 1990 to 2000 for the municipalities within the 50-mile radius of LaSalle is +37.3%. Using the 2000 50-mile radius population calculated in the next section, the 1990 50-mile radius population around LaSalle is calculated as follows:

1,553,566 persons / 1.373 = 1,131,512 persons

Summary of NUREG/CR-5305 50-mile Radius Population Estimation

The 50-mile radius population used in the LaSalle NUREG/CR-5305 consequence calculations is required to determine the current consequence estimates to be used in this ILRT risk assessment. As the NUREG/CR-5305 study does not report the 50-mile radius population, three methods have been used here to estimate the population used in the NUREG/CR-5305 study.

The best estimate of the 1990 population within 50 miles can be obtained by using the approximate growth rate for the specific area around LaSalle as determined from Table A-2 which is based on the 1990 and 2000 census.

The best estimate of these three approaches for the 1990 population within 50 miles of LaSalle is judged to be 1,131,512 persons. The value of 1,131,512 persons is used in this risk assessment as the NUREG/CR-5305 50-mile radius population.

Table A-2

2000 CENSUS POPULATION COMPARED TO 1990 FOR MUNICIPALITIES WITHIN 50 MILE RADIUS OF THE LASALLE SITE[1]

(Source: US Census 2000 Redistricting Data Summary File, PL 94-171)

Municipality	2000 Census Total Population	1990 Census Total Population	1990-2000 Population Change	1990-2000 % Change
Aurora city	142,990	99,581	43,409	43.6%
Naperville city	128,358	85,351	43,007	50.4%
Joliet city	106,221	76,836	29,385	38.2%
Bolingbrook village	56,321	40,843	15,478	37.9%
DeKalb city	39,018	34,925	4,093	11.7%
Woodridge village	30,934	26,256	4,678	17.8%
Kankakee city	27,491	27,575	(84)	-0.3%
Batavia city	23,866	17,076	6,790	39.8%
Lisle village	21,182	19,512	1,670	8.6%
Romeoville village	21,153	14,074	7,079	50.3%
Geneva city	19,515	12,617	6,898	54.7%
Ottawa city	18,307	17,451	856	4.9%
New Lenox village	17,771	9,627	8,144	84.6%
Bourbonnais village	15,256	13,934	1,322	9.5%
Lockport city	15,191	9,401	5,790	61.6%
Mokena village	14,583	6,128	8,455	138.0%
Streator city	14,190	14,121	69	0.5%
Crest Hill city	13,329	10,643	2,686	25.2%
Oswego village	13,326	3,876	9,450	243.8%
Lemont village	13,098	7,348	5,750	78.3%
Plainfield village	13,038	4,557	8,481	186.1%
Sycamore city	12,020	9,708	2,312	23.8%
Morris city	11,928	10,270	1,658	16.1%
Pontiac city	11,864	11,428	436	3.8%

[1] The municipalities used in this growth rate determination represent the majority of the city population within 50 miles of the LaSalle plant.

Table A-2

2000 CENSUS POPULATION COMPARED TO 1990 FOR MUNICIPALITIES WITHIN 50 MILE RADIUS OF THE LASALLE SITE[1]

(Source: US Census 2000 Redistricting Data Summary File, PL 94-171)

Municipality	2000 Census Total Population	1990 Census Total Population	1990-2000 Population Change	1990-2000 % Change
North Aurora village	10,585	5,940	4,645	78.2%
Frankfort village	10,391	7,180	3,211	44.7%
Marseilles city	4,655	4,811	(156)	-3.2%
Seneca village	2,053	1,878	175	9.3%
Grand Ridge village	546	560	(14)	-2.5%
Ransom village	409	438	(29)	-6.6%
Verona village	257	242	15	6.2%
Kinsman village	109	112	(3)	-2.7%
TOTALS:	829,955	604,299	225,656	37.3%

[1] The municipalities used in this growth rate determination represent the majority of the city population within 50 miles of the LaSalle plant.

A.2 YEAR 2000 50-MILE RADIUS POPULATION AROUND LASALLE

A calculation of the 2000 50-mile radius population around LaSalle was performed in support of this risk assessment. The calculation is documented in Exelon RM Documentation No. 843. [22]

Calculation RM No. 843 used 2000 Census data, as reported by the US Census Bureau on the web site http://quickfacts.census.gov/qfd/states/17000.html, along with Illinois maps to perform the population estimation.

The LaSalle plant is located in the town of Marseilles in LaSalle County, Illinois. The location of the site and the 50-mile radius is illustrated in Figure A-2 (Figure A-2 is an illustration for discussion purposes – more detailed maps were used in Calculation RM No. 843 to apportion populations). If the entire county falls within the 50-mile radius, based on a review of a map containing a mileage scale and county borders, then the entire population was included in the population estimate. Otherwise, a fraction of the population was counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius was estimated based on visual inspection of the map and the population of that area was estimated assuming uniform distribution of the population within the county.

Five counties were completely inside the fifty-mile radius. For the other counties, their percentage included in the fifty-mile radius was estimated and then multiplied by their total population. Since the population densities within some counties varied greatly, exceptions were made for the following counties: McLean, Kankakee, DeKalb, Cook, Lee, and Will.

Figure A-3

ILLUSTRATION OF 50-MILE RADIUS AROUND LASALLE SITE

McLean County: The fifty-mile radius does not include the cities of Bloomington and Normal with populations of 64,808 and 45,586, respectively (www.suntimes.com/census/cities/). The population of those cities was subtracted from the total population of McLean County then multiplied by 40% for a more accurate count.

Lee County: The only area densely populated is the city of Dixon, which is not included in the fifty-mile radius. The population of Dixon (15,941) was subtracted from the total population of Lee County before multiplying that figure by 60%.

Kankakee County: The major cities of Kankakee, Bradley, and Bourbonnais (27,491, 12,784, and 15,256, respectively) were all included inside the fifty-mile radius in the county of Kankakee, so the total population was multiplied by a higher percentage, 80%.

Dekalb County: The large cities of DeKalb and Sycamore were both included inside the fifty-mile radius in DeKalb County. DeKalb's population not including those two cities was multiplied by 70% and then added to DeKalb and Sycamore's total population.

Cook County: The small portion of Cook County included inside the fifty-mile radius was comprised almost completely of the town, Romeoville. The population of Romeoville (21,153) was used for the Cook County population estimate.

Will County: All major cities were included within the 50 mile zone. The area within the zone was adjusted from 80% to 90% to account for the higher density within the zone.

Based on Exelon RM Documentation No. 843, the total year 2000 population within a 50-mile radius of LaSalle Nuclear Station is estimated at 1,553,566 persons.

Appendix C
EXTERNAL EVENT ASSESSMENT

C.1 INTRODUCTION

This appendix discusses the external events assessment in support of the LaSalle ILRT frequency extension risk assessment. This appendix uses as the starting point of this assessment the external event work documented in the LaSalle EDG Completion Time risk application. [C-1]

Background

Exelon[1] submitted the results of the RMIEP study (NUREG/CR-4832) to the NRC in 1994 as the basis for the LaSalle IPE/IPEEE Submittal. Each of the RMIEP external event evaluations were reviewed as part of the Submittal and compared to the requirements of NUREG-1407. The NRC transmitted to Exelon in 1996 their Staff Evaluation Report of the LaSalle IPE/IPEEE Submittal. No other LaSalle external event PSA models or analysis were developed by Exelon.

C.2 EXTERNAL EVENT SCREENING ASSESSMENT

The purpose of this portion of the assessment is to examine the spectrum of possible external event challenges to determine which external event hazards should be explicitly addressed as part of the LaSalle ILRT frequency extension risk assessment.

Volume 7 of NUREG/CR-4832 provides the LaSalle RMIEP external event screening analysis. The screening assessment appropriately begins with the comprehensive list of potential external event hazards provided in the PRA Procedures Guide, NUREG/CR-2300. Consistent with NUREG/CR-2300, the screening assessment employed the following criteria to eliminate external event challenges from further consideration:

---

[1] Formerly ComEd.

1. The event is of equal or lesser damage potential than the events for which the plant is designed, or
2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events, or
3. The event cannot occur close enough to the plant to affect it, or
4. The event is included in the definition of another event

Although not listed explicitly as one of the screening criteria, the RMIEP screening assessment does incorporate (as evidenced in the Table 3.2-1 of Volume 7) the following criterion employed in the NUREG/CR-4550 study: "The event is slow in developing and there is sufficient time to eliminate the source of the threat or to provide an adequate response." This criterion is also considered appropriate.

Aside from seismic and internal fires (which are identified specifically as part of Generic Letter 88-20, Supplement 4), the following external events were identified in the RMIEP screening assessment for further analysis:

- Aircraft Impact
- Extreme Winds and Tornadoes
- Transportation/Toxic Chemicals/Explosions
- Turbine Generated Missiles
- External Flooding

Further assessment of each of these hazards is discussed below.

Seismic

Consistent with Generic Letter 88-20, the RMIEP study and the LaSalle IPEEE Submittal do not screen out this hazard but provide quantitative analyses. This is appropriate. This hazard is maintained in this assessment for further consideration.

Internal Fires

Consistent with Generic Letter 88-20, the RMIEP study and the LaSalle IPEEE Submittal do not screen out this hazard but provide quantitative analyses. This is appropriate. This hazard is maintained in this assessment for further consideration.

Aircraft Impact

Section 3.4.2 of Volume 7 of the RMIEP study provides a bounding assessment of the aircraft impact hazard. The assessment approach is consistent with the guidance provided in NUREG/CR-5042, Evaluation of External Hazards to Nuclear Power Plants in the United States, (identified in Generic Letter 88-20 as a source of acceptable methods to be used in the assessment of projected low frequency external events).

The LaSalle RMIEP bounding assessment conservatively assumes that any impact to a Category I structure sufficient to cause back face scabbing of an exterior wall results in a core damage probability of 1.0. The resulting bounding core damage frequency was estimated at 4.84E-7/yr.

The LaSalle RMIEP bounding assessment did not include the diesel generator building in the assessment because it is much smaller than the other key buildings and it is shielded on two sides by other buildings. Using the RMIEP-calculated reactor building aircraft impact CDF contribution of 3.93E-7/yr (obtained from Table 3.4-5 of NUREG/CR-4832 Volume 7), the contribution from an aircraft impact on the diesel generator building is estimated here as follows:

$$3.93E\text{-}7/yr \times 0.20 \times 0.50 \times 1.00 = 3.93E\text{-}8/yr$$

where:

0.20 = DG Bldg. area / Rx Bldg. area (based on review of M dwgs)

0.50 = 2 of the 4 compass directions are protected by other buildings

1.00 = Per the RMIEP assumptions, the CCDP is 1.0

Incorporating the DG building into the RMIEP bounding assessment framework results in a conservative CDF estimate of 5.23E-7/yr due to aircraft impacts.

If it is assumed here that an aircraft impact sufficient to result in back face scabbing of building exterior walls does not conservatively result in a CCDP of 1.0 (as assumed in the RMIEP framework), but rather a more reasonable value on the order of 0.1 or less, the aircraft impact induced CDF is estimated in the mid to lower E-8/yr range. Such an estimate is less than 1% of the LaSalle Revision 2003A PSA CDF. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to this assessment; therefore, such sequences are appropriately excluded from further analysis.

Extreme Winds and Tornadoes

Section 3.4.3 of Volume 7 of the RMIEP study provides a bounding assessment of extreme wind and tornado hazards. The assessment considers the pressure loading of extreme winds and tornadoes on both seismic Category I and non-Category I structures, failure of non-Category I structures onto Category I structures, and the effects of tornado generated missiles . The LaSalle Category I structures are designed to the following Design Basis Tornado (DBT) loadings:

- maximum rotation velocity of 300 mph
- transnational velocity of 60 mph
- external pressure drop of 3 psi
- impacts from postulated tornado missiles (e.g., wood plank, automobile)

The non-Category I structures are designed to withstand 90 mph straight winds.

As the LaSalle Category I structures are designed to 300 mph winds, the RMIEP study determined the frequency of wind pressure induced failures of Category I buildings to be negligible (<1E-6/yr). With respect to tornado-generated missiles, the study concluded that deformable and non-deformable missiles are not significant contributors

to plant risk (e.g., the contribution to plant risk due to the automobile missile impact on a Category I structure was estimated at less than 1E-8/yr). In addition, building air intakes and exhausts are protected from missiles by concrete barriers. Also, the ventilation stack is designed to withstand the effects of the DBT and therefore will collapse (onto the Auxiliary Bldg.) with a very low probability.

The plant risk contribution from extreme wind and tornado effects on non-Category I structures was estimated in the 1E-8/yr range. Although these buildings are more easily damaged, they do not contain equipment necessary for safe shutdown.

Due to the design of the LaSalle plant, the effect of extreme winds and tornadoes on plant safe shutdown is characteristic of LOOP and DLOOP initiator challenges.

The RMIEP study concluded that the median core damage frequency contribution from extreme wind and tornado hazards is 3E-8/yr. Although not specifically listed in the RMIEP study, the mean value is estimated here at 7.5E-8/yr (assuming a lognormal distribution and an error factor of 10). This estimate is approximately 1% of the LaSalle Revision 2003A PSA base CDF, and approximately 5% of the LOOP/DLOOP-initiated CDF. The tornado impact on LOOP/DLOOP accident sequences is already incorporated into the LOOP and DLOOP initiating frequencies and the LOOP and DLOOP offsite AC power recovery probabilities. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT frequency extension risk assessment. Such sequences are judged appropriately subsumed into the existing "internal events" analysis and are therefore excluded from further analysis.

Transportation

Section 3.4.4 of Volume 7 of the RMIEP study provides a bounding assessment of transportation hazards. The assessment addresses the frequency of occurrence of transportation accidents and the fragility of the plant to the associated effects (i.e., explosion forces, and toxic chemicals).

The maximum probable explosion hazard is a truck accident on nearby County Road 6 (6 miles south of the plant) involving an explosive force equivalent to a 50,000 lb. load of TNT. The walls of all LaSalle safety-related structures are designed to a minimum loading capacity of 3.0 psi. Using a conservative modeling approach documented in NUREG/CR-2462, the lower bound capacity of structural panels at LaSalle was conservatively estimated at 1.95 psi. Comparison of this calculated minimum wall capacity to the free-field incident overpressure of 0.66 psi due to the truck blast, shows that at least a factor of 3 capacity exists against the blast loading. The RMIEP study appropriately concluded that explosions due to transportation accidents are a negligible contributor to plant risk.

Regarding toxic chemical releases, the RMIEP study reviewed the types and amounts of chemicals typically stored and transported in and around the LaSalle site. Among the three transportation modes near the site, a barge accident in the Illinois River could result in the largest amount of chemical spill. The Illinois River is 3.5 miles away from the plant structures at its closest distance. Also, the river elevation is approximately 180 feet below the plant grade. Given that many toxic vapors are denser than air, the atmospheric dispersion of these chemicals towards the plant under favorable wind conditions is unlikely because of the difference in plant and river elevations. Also, for more turbulent wind conditions, it is highly unlikely that a toxic vapor would reach the control room air intakes at excessive concentrations. The RMIEP study appropriately concluded that toxic chemical releases are negligible contributors to plant risk.

Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment.

Turbine Missiles

Section 3.4.5 of Volume 7 of the RMIEP study provides a bounding assessment of turbine missile hazards. The RMIEP assessment estimates the frequency of turbine missile induced core damage at less than 1E-7/yr and concludes that the hazard is not a significant contributor to risk. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment; therefore, such sequences are appropriately excluded from further analysis.

External Flooding

Section 3.4.6 of Volume 7 of the RMIEP study provides a bounding assessment of the external flooding hazard. The assessment appropriately considers the following three external flooding sources:

- Nearby Illinois River
- LaSalle cooling lake
- Local precipitation

The plant grade level is at 710' mean sea level (MSL). All safety-related structures at the LaSalle station have a ground floor surface elevation of at least 710.5' (MSL). An inspection of the plant was made as part of the RMIEP study. The inspection revealed that ground floor doors are leak tight; even if external water levels were to rise above plant grade, the buildings would not be flooded.

The probable maximum flood elevation of the Illinois River, including coincident wave effect, is 522.5'. This level is 188 feet below the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures. Failures of low navigation dams existing upstream of the plant would also not affect the site.

The cooling lake is at a lower elevation, 700' MSL, than the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures. Runoff from the lake (due to intense precipitation or breaching of the lake dikes) would flow away from the cooling lake into local creeks that meet the Illinois River.

The probable maximum precipitation (based on conservative assumptions) is calculated to result in a water level elevation at the LaSalle site of approximately 710.3' MSL, slightly lower than the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures.

The RMIEP study appropriately excludes external flood hazards as negligible contributors to plant risk. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment; therefore, such sequences are appropriately excluded from further analysis.

Conclusions of Screening Assessment

Given the foregoing discussions, the following external event hazards are judged not screened out and are evaluated further in the LaSalle ILRT Extension Submittal:

- Seismic events
- Internal fires

The other external hazards are assessed to be negligible contributors to plant risk. Explicit treatment of these other external hazards is not necessary for most PSA applications (including the ILRT Extension Submittal) and would not provide additional risk-informed insights for decision making.

C.3 SEISMIC ASSESSMENT

Seismic induced accident sequences are included in the LaSalle PSA Revision 2003A (i.e. the current model of record, and the PSA models used in this ILRT risk assessment). The seismic sequences in the LaSalle model of record are based on rigorous seismic PRA work performed for the LaSalle RMIEP study.

This section discusses the seismic induced accident sequence assessment.

C.3.1 RMIEP Seismic Overview

The RMIEP study analyzed LaSalle seismic risk employing the methodology sponsored by the U.S. NRC under the Seismic Safety Margin Research Program (SSMRP) and developed by Lawrence Livermore National Laboratory (LLNL). The key elements of the LaSalle RMIEP seismic risk analysis are:

1. Development of the seismic hazard at the LaSalle site including the effect of local site conditions.
2. Comparisons of the best estimate seismic response of structures, components, and piping systems with design values for the purposes of specifying median responses in the seismic risk calculations.
3. Investigation of the effects of hydrodynamic loads on seismic risk.
4. Development of building and component fragilities for important structures and components.
5. Development of the system models (e.g., event and fault trees).
6. Estimation of the seismically induced core damage frequency.

This approach to seismic risk assessment is consistent with the requirements of the NRC IPEEE Program and current seismic risk assessment technology. Overviews of these elements are provided below.

RMIEP Seismic Hazard Frequency

The LaSalle seismic hazard curve used in the RMIEP study is based on the NRC sponsored Eastern United States Seismic Hazard Characterization study (NUREG/CR-5250) performed by Lawrence Livermore National Laboratory (LLNL) in the 1980's. The LaSalle RMIEP hazard curve is divided into seven discrete seismic magnitude ranges for final sequence quantification:

- LL1: magnitude 0.10-0.18g
- L1: magnitude 0.18-0.27g
- L2: magnitude 0.27-0.36g
- L3: magnitude 0.36-0.46g
- L4: magnitude 0.46-0.58g
- L5: magnitude 0.58-0.73g
- L6: magnitude >0.73g

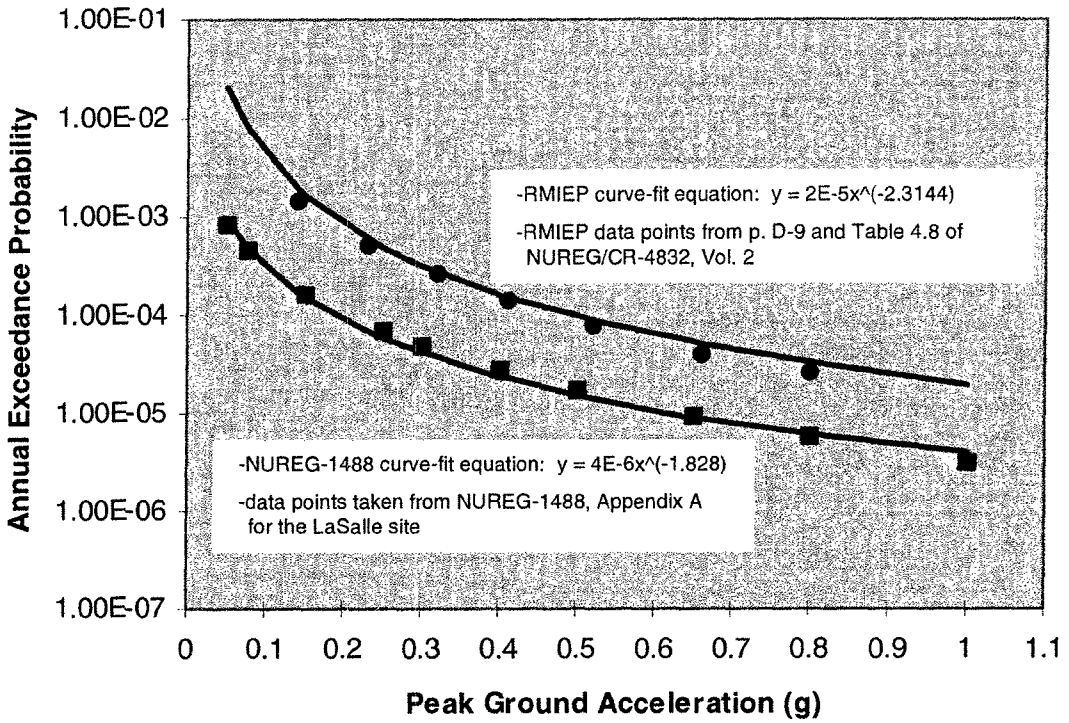
The LLNL seismic hazard curves used in the RMIEP study are more conservative than the latest NRC estimates and the EPRI estimates. In conjunction with providing funding to LLNL in the 1980's to perform a probabilistic seismic hazard analysis (PSHA) study, the NRC recommended that the nuclear power industry perform an independent study to provide the NRC with comparative information. A consortium of nuclear power utilities funded EPRI to perform a seismic hazard study. EPRI developed its own PSHA methodology and PSHA estimates at 56 of the eastern United States sites (documented in EPRI NP-4726 and EPRI NP-6395D). The differences between the 1980's LLNL and the EPRI seismic hazard estimates (the EPRI curves were generally lower) are addressed in NUREG/CR-4885.

During 1992 and 1993, LLNL re-elicited input data from their seismicity and ground motion experts using a revised elicitation procedure. LLNL then revised their PSHA computer code and produced updated PSHA estimates at eastern United States sites. The updated LLNL methodology reduced the seismic hazard estimates below that of the 1980's study, thus reducing the differences between the LLNL and EPRI hazard estimates. According to NUREG-1488, the updated LLNL seismic hazard estimates will be considered by the NRC staff in future licensing actions such as safety evaluation reports, reviews of individual plant examination of external events (IPEEE) submittals, and early site reviews.

The seismic hazard curve used in the LaSalle RMIEP study is compared with the latest NRC estimates (taken from NUREG-1488) in Figure C.3-1. As can be seen from Figure C.3-1, the hazard frequencies used in the RMIEP study are approximately a factor of 5 higher than those assessed using the latest NRC estimates.

Figure C.3-1

COMPARISON OF LASALLE RMIEP AND
NUREG-1488 SEISMIC HAZARD CURVES



NOTES:

1. RMIEP study seismic hazard curve: circle data points
   NUREG-1488 LaSalle site seismic hazard curve: square data points

2. RMIEP data points are plotted as the middle pga value of the discrete RMIEP seismic level range (the middle pga value for the >0.73g range is estimated here as 0.8g) with the mean frequency from Table 4.8 and page D-9 of NUREG/CR-4832, Volume 2.

3. Smooth curves are Microsoft Excel curve-fits to the RMIEP study and NUREG-1488 discrete data points (see chart text for equations).

RMIEP Seismic Response Analysis

Seismic responses, together with fragilities, allow for the calculation of seismically induced failure probabilities. The seismic response task generated probabilistic seismic responses for all structures and equipment identified in the PSA models. The SMACS methodology (NUREG/CR-2015) of the SSMRP was used in the LaSalle RMIEP response analysis. SMACS analyses were performed on LaSalle structures, including the effects of soil-structure interaction (SSI). SMACS links together seismic input, SSI, structure response, and piping system and component response.

RMIEP Hydrodynamic Load Investigation and Load Combination Approach

The RMIEP study evaluated the probabilities of failure of a particular structure or equipment due to earthquake occurrence by including the effect of the hydrodynamic loads which may occur concurrently with the earthquake. The hydrodynamic loads identified and considered in the RMIEP analysis are: safety/relief valve discharge loads, LOCA-induced loads, jet forces, pool swell, condensation-oscillation (CO), and chugging. It was determined that hydrodynamic loads which may be experienced in BWRs during an earthquake are not significant at LaSalle.

RMIEP Fragility Analysis

The RMIEP structural fragility analysis followed the SSMRP structural fragility assessment methodology as documented in NUREG/CR-2320. Detailed fragility assessments were performed for various shear walls and diaphragms, the primary containment, and concrete members inside containment. Structural fragilities were assessed in terms of equivalent elastic capacities.

The RMIEP equipment fragility analysis followed the SSMRP subsystem fragility assessment methodology as documented in NUREG/CR-2405. Fragilities for selected LaSalle components were derived by extrapolating design information. The fragilities are defined as the conditional probability of failure given a specified structural response.

The equipment fragilities are assumed to fit a lognormal distribution and are defined by a spectral acceleration capacity and two randomness and uncertainty variables. LaSalle specific fragilities were assessed for approximately three dozen key components, subsystems, and component types. Generic fragilities for other equipment were obtained from available industry studies.

The RMEIP general conclusion regarding this aspect of the seismic analysis is that the LaSalle plant is designed very well from a seismic point of view. Seismic induced structural and equipment failures, other than loss of offsite power (refer to Table C.3-1), do not contribute significantly to LaSalle seismic risk.

RMIEP Seismic PSA Models

The RMIEP study considers the following potential seismic induced accident sequence initiating events:

Seismic-Induced Initiator	Assessment
RPV Rupture	Not significant likelihood; no sequences explicitly modeled
ISLOCA/BOC	Not significant likelihood; no sequences explicitly modeled
LLOCA	3+ SORVs following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled
MLOCA	2 SORVs following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled
SLOCA	1 SORV following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled

Seismic-Induced Initiator	Assessment
Transient	Loss of Offsite Power likely for most seismic events. Loss of offsite power subsumes all other potential transients. Sequences explicitly modeled.

Table C.3-1

OFFSITE POWER FRAGILITIES (RMIEP)

RMIEP Event	Description	Mean Value
LOSP-LL1	Loss of Offsite Power due to ceramic insulator failure in switchyard from LL1 seismic initiator	2.48E-01
LOSP-L1	Loss of Offsite Power due to ceramic insulator failure in switchyard from L1 seismic initiator	2.95E-01
LOSP-L2	Loss of Offsite Power due to ceramic insulator failure in switchyard from L2 seismic initiator	3.71E-01
LOSP-L3	Loss of Offsite Power due to ceramic insulator failure in switchyard from L3 seismic initiator	4.36E-01
LOSP-L4	Loss of Offsite Power due to ceramic insulator failure in switchyard from L4 seismic initiator	5.00E-01
LOSP-L5	Loss of Offsite Power due to ceramic insulator failure in switchyard from L5 seismic initiator	5.75E-01
LOSP-L6	Loss of Offsite Power due to ceramic insulator failure in switchyard from L6 seismic initiator	6.59E-01

The RMIEP event tree structure for seismic events is taken directly from the RMIEP internal event trees. Any event in the fault tree which could be the result of either a random failure or a seismically induced failure was modified by adding OR-gates with two basic event inputs. After the event trees and fault trees were developed, a detailed database providing the basic events, associated response fragility, and random failure data was generated to feed into the SEISIM code to yield the CDFs for all earthquake levels.

The following key assumptions and modeling issues are incorporated into the RMIEP seismic accident sequence structure:

- Seismic events that do not trigger seismic-induced loss of offsite power are not explicitly modeled, they are assessed as not risk significant.
- All modeled seismic sequences involve loss of offsite power, as such, systems dependent upon offsite power (e.g., Feedwater, Condensate, CRD, power conversion, etc.) are not modeled.
- Offsite AC power recovery is assigned a failure probability of 1.0 for all seismic levels.
- Onsite AC power recovery is credited, except in the case of common cause diesel generator failure.
- Primary containment venting is not credited.

RMIEP Seismic Quantification Results

The total seismic core damage frequency is estimated in the RMIEP study at a mean value of 7.58E-7/yr. More than 98% of the total seismic frequency is comprised of seismic induced station blackout sequences involving initial RCIC operation. Approximately 1% of the seismic CDF are seismic induced loss of offsite power sequences involving stuck open relief valves. The high percentage of station blackout core damage sequences is not surprising given that the RMIEP seismic sequences do not credit recovery of offsite power.

RMIEP Conclusions Regarding LaSalle Seismic Risk

The LaSalle seismic risk is dominated by seismic-induced loss of offsite power initiators followed by random equipment failures. The key conclusions of the RMIEP seismic analysis are best described by the following passages from NUREG/CR-4832, Volume 2, Section 4:

> *"The primary characteristic of the dominant sequences at LaSalle is that the only explicitly seismic events appearing in the final cut sets are the seismic initiating event frequencies for each level and the seismically induced loss of offsite power conditional probabilities at each level. No other seismic failures or seismic related events survived the initial and final quantifications. This is very different than the results for many other plants. The LaSalle plant is very well designed from a seismic view-point. The detailed structural analysis performed in Volume 8 did not find any structural failures where walls might fall and damage critical equipment, the cabinets and panels were bolted down correctly, and the piping penetrations were designed appropriately to handle any shifting as a result of the seismic event. The accident sequences, therefore, are equivalent to seismically induced transients.*
>
> *If a LOSP was not likely to occur as a result of the seismic event, there would be no dominant seismic sequences as LaSalle. No other seismically induced initiator has a significant conditional probability and compromises redundancy enough to result in accident sequences with a substantial frequency. The dominant sequences at LaSalle are, therefore, all seismically induced losses of offsite power except that no credit is given for recovering offsite power after the seismic failure."*

C.3.2 Seismic Modeling For LaSalle ILRT Extension Submittal

The LaSalle seismic analysis performed for the RMIEP study is a rigorous LaSalle specific analysis. The methodology used is consistent with the requirements of the NRC IPEEE Program and with current seismic risk assessment technology. The general conclusions regarding the seismic response of the LaSalle plant are judged still applicable. Specific dominant sequences and cutsets may currently differ due to plant procedural and PSA model changes. As the LaSalle seismic risk is sensitive to EDG availability and reliability, seismic sequences are explicitly included in the LaSalle PSA

model of record. No additional seismic PSA effort other than this discussion has been performed in support of this ILRT risk assessment.

The seismic modeling approach used in the LaSalle PSA is based on the general conclusions of the RMIEP study and is as follows :

- The division of the LaSalle seismic hazard curve into seven discrete seismic magnitude ranges is maintained in this assessment (the same ranges used in the RMIEP study are maintained).

- Instead of the 1980's vintage seismic initiator frequencies used in the RMIEP study, this assessment uses the more current NUREG-1488 based frequencies (refer to Figure C.3-1). These are:

Seismic Magnitude Range		Exceedance Frequency
LL1:	Magnitude 0.10 – 0.18g	2.7E-4/yr
L1:	Magnitude 0.18 – 0.27g	9.2E-5/yr
L2:	Magnitude 0.27 – 0.36g	4.4E-5/yr
L3:	Magnitude 0.36 – 0.46g	2.6E-5/yr
L4:	Magnitude 0.46 – 0.58g	1.7E-5/yr
L5:	Magnitude 0.58 – 0.73g	1.1E-5/yr
L6:	Magnitude > 0.73g	7.1E-6/yr

  These frequencies are conservatively taken at the beginning point of each magnitude range (e.g., the 2.7E-4/yr frequency for the LL1 range is calculated based on a 0.10 pga seismic event).

- The RMIEP loss of offsite power fragilities (refer to Table C.3-1) are judged reasonable and are maintained in this assessment.

- The seismic hazard frequencies and associated offsite power fragilities are combined into the following seismic event tree initiating events:

Initiator ID	Description	Frequency
%SEIS-LL1	LL1 Seismic–Induced DLOOP Event	6.7E-5/yr
%SEIS-L1	L1 Seismic–Induced DLOOP Event	2.7E-5/yr
%SEIS-L2	L2 Seismic–Induced DLOOP Event	1.6E-5/yr
%SEIS-L3	L3 Seismic–Induced DLOOP Event	1.1E-5/yr
%SEIS-L4	L4 Seismic–Induced DLOOP Event	8.5E-6/yr
%SEIS-L5	L5 Seismic–Induced DLOOP Event	6.3E-6/yr
%SEIS-L6	L6 Seismic–Induced DLOOP Event	4.7E-6/yr

- Each of the above seismic initiators is propagated through the accident sequence quantification of the base LaSalle model. These seismic sequences are characterized as follows:
  - The sequences are dual-unit LOOPs and the base LaSalle DLOOP event tree structure is employed.
  - Consistent with the insights of the RMIEP seismic study, the only seismic-induced equipment or structural failures explicitly modeled in this assessment are the offsite power insulators.
  - Offsite AC recovery is not credited.
  - Emergency diesel generator recovery is not credited, consistent with the base LaSalle model.
  - As these sequences are DLOOPs and offsite power recovery is not credited, systems dependent upon offsite power (e.g., Feedwater, Condensate, Containment Venting, etc.) are not available to support accident mitigation.
  - Alternate injection using the diesel fire pump is credited for long term accidents (i.e., accidents with initial RPV injection via another system such as RCIC).
- Consistent with the insights of the RMIEP seismic study, seismic-induced RPV Rupture, ISLOCA, LOCA (SORVs following the seismic-induced DLOOP initiators are modeled) and BOC sequences are not explicitly quantified because they are assessed as not significant contributors to seismic risk.

Base Seismic Sequence Quantification Results

The LaSalle base seismic-induced core damage frequency is estimated at 1.17E-7/yr.

The numerical difference between this seismic-induced CDF and that estimated by RMIEP (7.58E-7/yr) is appropriately explained by the following two key factors:

- Use of the more current NUREG-1488 seismic initiator frequencies (the RMIEP frequencies are approximately a factor of 5 higher)
- Refinements to the LaSalle PSA since the RMIEP study (including key contributors such as the reduction in EDG failure rates to reflect plant Maintenance Rule Program data).

The dominant accident sequence types are station blackout scenarios, which represent approximately 80% of the seismic CDF. The dominant cutsets are seismic-initiated DLOOP events with successful RCIC operation and common cause failure of the emergency diesel generators (which result in core damage in approximately 8-9 hours due to battery depletion at 7 hours). These results are consistent with those of the RMIEP study (74% of the RMIEP seismic CDF is represented by such cutsets).

C.4 INTERNAL FIRES ASSESSMENT

This internal fires assessment is based on the extensive work performed for the LaSalle RMIEP study.

C.4.1 RMIEP Internal Fires Overview

The internal fires LaSalle RMIEP study is a detailed analysis that, like the seismic analysis, uses quantification and model elements (e.g., system fault trees, event tree structures, random failure rates, common cause failures, etc.) consistent with those employed in the internal events portion of the RMEIP study. The LaSalle RMIEP internal fires study was performed during the same time frame as the NUREG-1150 studies and The Fire Risk Scoping Study.

The RMIEP internal events study models were used to support sequence quantification. This ensured that the fire sequence quantifications included plant-specific line-up, reliability, and human pre-accident reliability data. Plant walkdowns were performed to document plant-specific combustible loading, suitability of fire severity factors, locations of critical equipment, locations of fire dampers, suitability of doors and other fire barriers, effectiveness of fire detection and suppression systems, and other component specific attributes. Plant-specific cable location data were used to spatially identify control and power cables passing through or powering components in the various fire areas.

The key elements of the LaSalle RMIEP internal fires assessment are consistent with current approaches and include:

1. Fire hazard analysis
2. Fire growth and propagation
3. Fire suppression.
4. Accident sequence development and quantification.

Overviews of these elements are provided below.

Fire Hazard Analysis

The LaSalle RMIEP fire hazard analysis is typical of fire PRA techniques and involves dividing the plant into discrete fire areas, estimating fire ignition frequencies for each fire area, and identifying critical fire areas for detailed quantitative assessment.

The RMIEP study uses the Appendix R fire areas and zones as a starting point for defining discrete fire areas. These areas are modified to account for barriers and equipment separation within fire areas. This partitioning is based on review of plant equipment location and arrangement drawings, plant Fire Hazards Analysis (FHA) discussions, and plant walkdowns. Fire area boundary definitions are based on the following:

- NRC Generic Letter 83-33 (10/19/83) definition of a fire area
- engineering judgment
- available level of detail of cable and component location information

A detailed list of the identified fire areas, descriptions of areas and barriers, and the bases for the boundary assessments are provided in Tables 3.3 and 3.4 of NUREG/CR-4832, Volume 9. Of the 160 LaSalle FHA defined fire zones, 54 PSA fire areas were identified.

The RMIEP fire ignition frequencies are estimated based primarily on the fire events database provided in NUREG/CR-4586, Users' Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base (the database is compiled from information presented in NUREG/CR-5088, the Seabrook PSA, and the Limerick Severe Accident Risk Assessment). Fire area ignition frequencies are estimated for the following eight general plant buildings/areas: 1) control room; 2) cable spreading room; 3) diesel generator room; 4) electrical switchgear room; 5) battery room; 6) reactor building; 7) turbine building, and 8) auxiliary building. Estimation of specific fire area ignition frequencies is generally calculated as the ratio of the floor area in question to that of the larger building. In some cases, a specific fire area ignition frequency is based on the ratio of the foot-print area of the most probable ignition sources in a fire area (based on walkdown information) to that of the larger building.

To determine the fire areas warranting detailed quantification, the RMIEP study performs an initial screening quantification. The RMIEP internal events fault trees were used to identify all key components and cabling credited in the PSA. Plant schematics were used to map components to locations. Cables were identified from master electrical wiring diagrams. This information and Sargent and Lundy cable routing information for LaSalle were used to map fault tree basic events to associated equipment and cable locations.

The RMIEP internal event transient event tree structure is employed in the initial screening quantification of the fire areas. The fire ignition frequency of each fire area was set to 1.0 and all functions in the area were set to fail using the location information. In addition, a screening fire barrier failure rate of 0.1/demand was applied between fire areas in this initial screening quantification. The initial screening quantification resulted in identification of the following critical fire areas for further detailed quantitative analysis:

ID	Room Description
5C11-4	Diesel Generator Corridor
4D2	Cable Spreading Area
4D4	Electrical Equipment Room
4E2-1	Auxiliary Equipment Room (Main Area)
4E2-2	Auxiliary Equipment Room (Northwest Corner)
4F3	Aux. Bldg. Rad. Chemistry Offices
5B13-2	BOP Cable Area (North)
4E4-1	Cable Shaft Area of Div. 2 Ess. SWGR Room
4C1	Control Room
4E4	Div. 2 Ess. SWGR Room
4F2	Div. 1 Ess. SWGR Room

The details of the fire hazard analysis and initial screening quantification are discussed in Sections 3.1 - 3.5 of NUREG/CR-4832, Volume 9.

Fire Growth and Propagation

Discrete fire scenarios were modeled for the critical fire areas that survived the initial screening quantification. The COMPBRN fire growth code was used to model fire growth and fire-induced equipment damage. The RMIEP fire scenarios are generally modeled with two fire types:

- "Small fire", modeled as a 2 ft. diameter 1 gallon oil spill
- "Large fire", modeled as a 3 ft. diameter 10 gallon oil spill

This is a conservative treatment of fire modeling (i.e., compared with the techniques of the EPRI Fire PRA Implementation Guide) and may generally over estimate the fire-induced equipment damage in many areas (e.g., cable spreading room).

The cable damage threshold used was 662°F, and the cable insulation ignition temperature used was 932°F.

Fire propagation in cable trays and hot gas layer effects were treated where appropriate.

Zones of damage were then determined for each fire scenario. Dominant cutsets from the initial screening quantifications were used to identify dominant critical areas in each critical fire area. Using this information, the floor area in a given fire area in which fire-induced damage to equipment of interest to the PSA could occur was estimated.

In addition to the conservative selection of fire types, the RMIEP study employed the following conservative approaches when determining fire-induced equipment damage:

- Fire-induced failure of any Main Steam equipment is modeled as failure of MFW, Condensate, and the PCS
- Fire induced failure of any mode of RHR is modeled as failure of all modes of RHR

- Fire-induced failure of RHR and containment vent is modeled to also fail the PCS.

Fire Suppression

Automatic suppression, when present, and fire brigades were credited for fire scenarios during the time frame before the COMPBRN predicted time to fire-induced equipment damage.

A detailed analysis of manual fire suppression was performed in support of the RMIEP internal fire analysis. The RMIEP manual suppression analysis was supported by plant walkdowns, review of installed suppression system information, review of procedures and practices, and interviews with plant fire personnel. The manual suppression failure probabilities consider: time to detection, time to assemble and suit-up, time to respond to scene, time to set-up at scene, time to search for fire source location, time to control fire.

Credit for automatic suppression systems considered the detector and head spacing with respect to the fire location, as well as the time to fire-induced equipment damage. The RMIEP automatic suppression failure probabilities are generic industry values taken from the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide failure rates based on five different industry sources: Water (3.8E-2 failure probability); Halon (5.9E-2 failure probability); and CO2 (4.0E-2). The NUREG-1150 automatic suppression system failure probabilities are generally consistent with the values provided in the EPRI FIVE Methodology, these are: Preaction and Deluge Systems (5.0E-2); Sprinkler Systems (2.0E-2); Halon (5.0E-2); and CO2 (4.0E-2).

Accident Sequence Development and Quantification

Each fire scenario that indicated potential fire-induced damage to equipment of interest to the PSA was modeled probabilistically and addressed the following issues:

- building fire ignition frequency
- area ratio of fire area to that of building
- area ratio within fire area where fire scenario results in damage to equipment
- fire severity ratio
- failure of automatic suppression systems
- failure of manual suppression
- random and fire-induced equipment failures

Fire-induced equipment failures were modeled by failing appropriate basic events in the PSA. The fire scenarios were then modeled with the internal events transient accident sequences to quantify the fire-induced core damage frequency for each scenario.

RMIEP Internal Fires Quantification Results

The total fire-induced core damage frequency was estimated in the RMIEP study at a mean value of 3.21E-5/yr. A summary of the RMIEP internal fires modeling and quantification is provided in Table C.4-2.

Consistent with other BWR internal fire PSAs, the dominant fire areas are the Control Room and the Essential Switchgear Rooms.

In all fire areas, additional (i.e., in addition to fire-induced equipment failures) random failures and/or operator errors are necessary to result in a core damage accident. In the case of the Control Room, the dominant scenario (consistent with other fire PSAs) is smoke-induced abandonment of the Control Room and failure to successfully control the plant from the remote shutdown panel.

Excluding the Control Room fire scenario, the majority (99%) of the RMIEP fire-induced core damage accidents are long-term loss of containment heat removal scenarios (Class II). The Control Room fire scenario is conservatively assumed in the RMIEP

study to result in a short term high-pressure loss of coolant injection accident (Class IA). In addition, the RMIEP fire analysis included a conservative evaluation of the Control Room fire frequency leading to core damage. Recent Exelon control room fire analyses indicate these conservative analyses are approximately a factor of ten too high in their CDF impact. Using a more realistic evaluation of the control room fire CDF results in the following evaluation of the accident break down for fire risk contributors:

- Class II = 90.9%
- Class IA = 8.1%
- Class ID = 1%

The fire-induced core damage frequency estimated for LaSalle in the RMIEP study is at the conservative end of the spectrum for the following reasons:

- The fire-induced damage indicated by the RMIEP fire scenario assessments are known to be conservative (i.e., the RMIEP assessment conservatively failed entire functions given fire induced failure of a portion of a system or of a related system).
- The RMIEP internal fire assessment conservatively assumes that each identified fire scenario represents 100% of the room ignition frequency.
- The Fire Severity factors used in RMIEP are generally conservative when compared to the EPRI Fire PRA Procedures Guide.

C.4.2 Application of RMIEP Internal Fire PSA to LaSalle ILRT Extension

As discussed in the previous section, the RMIEP calculated internal fires induced CDF is a conservative estimate. However, the qualitative conclusions of the RMIEP internal fires assessment are judged still applicable, though specific dominant sequences and cutsets may differ due to plant procedural and PSA model changes.

The LaSalle fire risk is dominated by long term core damage accidents. However, the LERF risk impact due to ILRT frequency changes is dominated by short term core damage accidents. As such, the explicit inclusion of internal fire accident sequences frequency information in this ILRT risk assessment would not significantly alter the

quantitative results nor would it change the conclusions of this analysis (i.e., the risk impact of ILRT interval extension to 15 years 15 months is very small). The change in LERF remains below 1E-7/yr.

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
(E) 5C11-4	Diesel Generator Corridor	• 241X (CW pumps 2A & C; PSW pumps 2A & 0; SA comp. 2SA01C; MCCs 231X, 231Y, 237X, 237Y) • 232Y-2 (FW pump 2B valves; RHR A service water strainer) • 232B-1 (alt. feed RPS buses A&B) • 125VDC Battery 2A (train A systems) • *Offsite Power*	None	Very large floor fire (10)	8-9	RCIC, MFW, Condensate, PCS, all LPCS, all RHR	3.36E-2	0.0038	0.30	0.17	1.0	0.83	1E-1 to 2E-1	6.20E-7	1.9
(N) 4D2	Cable *Spreading Area*	• Cables for train B *systems*	Auto *Sprinkler*	Large floor *fire (8)*	3-5	All train B safety *systems*, MFW, Cond., PCS, venting	6.48E-3	1.0	0.15	0.30	0.038	0.99	8E-3 to 2E-2	1.63E-7	0.5
(P) 4D4	Electrical Equipment Room	• RPS 120VAC Bus A • MG Set A • RPS 120VAC Bus B • MG Set B • MSIV Closure signal • *Train A system cables*	None	Large floor fire	7-8	All train A safety systems, MFW, Condensate, PCS, and venting	4.90E-2	0.06	0.05	0.30	1.0	0.97	7E-3 to 2E-2	3.28E-7	1.0
				Small floor fire	3-4	Same fire-induced *damage as for the* large floor fire			0.016	0.70	1.0	0.97	7E-3 to 2E-2	2.45E-7	0.8
(S) 4E2-1	Auxiliary Equipment *Room (Main* Area)	• Cables for train B systems	None	Large floor fire #S-AA *(8)*	4-5	Same as for (AA) 5B13-2: Offsite *power and venting*	4.90E-2	0.028	0.11	0.30	1.0	0.97	1E-4 to 2E-4	5.94E-9	0.0
				Large floor fire #S-W *(8)*	4-5	Same as for (W) 4E4: all train B *safety systems*, MFW, Cond., PCS, venting			0.11	0.30	1.0	0.97	8E-3 to 2E-2	3.52E-7	1.1

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
(T) 4E2-2	Auxiliary Equipment Room (Northwest Corner)	• Cables for train A systems	None	Large floor fire (8)	9-10	All train A safety systems, MFW, Cond., and PCS	4.90E-2	0.068	0.84	0.30	1.0	0.97	~3E-3	2.27E-6	7.1
(Z) 4F3	Aux. Bldg. Rad. Chem. Offices	• Cables for train A systems	Partial Sprinkler Coverage	Large floor fire (8)	5-6	All train A safety systems, MFW, Cond., PCS, venting	4.90E-2	0.082	0.005	0.30	1.0	0.91	7E-3 to 2E-2	3.58E-8	0.1
(AA) 5B13-2	BOP Cable Area (North)	• 242X (CW pump 2B; PSW pump 2B and jockey 0B; MCCs 232X, 232Y, 238) • Cables for train A systems	None	Large floor fire (8)	6-8	Offsite power and venting	4.90E-2	0.064	0.08	0.30	1.0	0.93	1E-4 to 2E-4	7.31E-9	0.0
(AC) 4E4-1	Cable Shaft Area of Div. 2 Ess. SWGR Room	• Cables for train B systems (11)	None	Small floor fire (9)	2-3	All train A safety systems (11), MFW, Cond., PCS, venting	4.90E-2	0.0016	1.0	1.0	1.0	0.99	7E-3 to 2E-2	5.42E-7	1.7
(G) 4C1	Control Room	• Cables for train A, B and C systems	None	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	1.39E-5 (1.39E-6)	43.3
(W) 4E4	Div. 2 Ess. SWGR Room	• 252 (train B non-safety AC) • 242Y (train B safety AC) • 236X (DGCWP 2A; RHRSW pump 2C) • 236X-2 (WW vent; MG Set B) • 236X-3 (125VDC train B charging) • 236Y (RHRSW pump 2D; RHR train B and C; DW vent; RCIC & RBCCW isolations; FW turbines; SLC train B)	None	Switchgear cubicle fire	4-5	All train B safety systems, MFW, Condensate, PCS, venting	7.97E-3	1.0	1.0	0.01 (13)	1.0	0.98	8E-3 to 2E-2	1.80E-6	5.6

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
		• 125VDC 2B Battery, Bus and charger (train B systems) • 125VDC 212X (FW pump 2B, DC to non-safety train B systems) • 125VDC 212Y (ADS train B, DC to train B safety systems)		Large floor fire (8)	4-5	Same fire-induced damage as for the SWGR cubicle fire			0.18	0.30	1.0	0.98	8E-3 to 2E-2	6.71E-6	20.9
(Y) 4F2	Div. 1 Ess. SWGR Room	• 251 (train A non-safety AC) • 241Y (train A safety AC) • 235X (DGCWP 0; RHRSW pump 2A; WW vent; RCIC & SDC isolations) • 235X-2 (MG Set A) • 235X-3 (125VDC train A and 250VDC charging) • 235Y (RHRSW pump 2B; RHR train A; LPCS; DW vent; SLC train A) • 125VDC 2A Bus and charger (train B systems) • 125VDC 211X (DC to non-safety train A systems) • 125VDC 211Y (DC to train A safety systems) • 250VDC 2 Battery, Bus and charger (RCIC, all 250VDC)	None	Switchgear cubicle fire	4-5	All train A safety systems, MFW, Condensate, PCS, venting	7.97E-3	1.0	1.0	0.01 (13)	1.0	0.95	7E-3 to 2E-2	1.76E-6	5.5
				Large floor fire (8)	4-5	Same fire-induced damage as for the SWGR cubicle fire			0.13	0.30	1.0	0.95	7E-3 to 2E-2	3.39E-6	10.6

Notes to Table C.4-2:

1) Deterministic fire modeling was performed using COMPBRN. The RMIEP study modeled fires with two general fire scenarios, a "small" 1 gallon oil fire and a "large" 10 gallon oil fire. This is a conservative treatment of fire modeling and may generally over estimate the fire-induced equipment damage in many areas (such as a cable spreading room). In addition, the RMIEP study made the following additional conservative assumptions when modeling fire-induced equipment failures: 1) fire induced failure of any main steam equipment was modeled as failure of MFW, Condensate and the PCS; 2) fire induced failure of one mode of RHR was modeled as failing all modes of RHR; and 3) modeling fire induced failure of RHR and Vent was extrapolated to also imply failure of the PCS. These lists of fire-induced equipment failures by fire scenario are based on review of cutsets and text discussions in the RMIEP internal fire analysis documentation (NUREG/CR-4832, Vol. 9).

2) The RMIEP fire ignition frequencies are based on the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide a compilation of fire events by eight key plant buildings/areas. The data is complied from information presented in NUREG/CR-5088, the Seabrook PSA, and the Limerick Severe Accident Risk Assessment.

3) The Fire Room to Fire Area ratio is a ratio of the floor area of the fire room to that of the larger fire area, and is used to partition the fire area ignition frequency to apply to the fire room in question.

4) The Fire Scenario to Fire Room ratio is a ratio of the floor area within the fire room in question where the fire scenario in question may be located and cause the damage of interest.

5) The RMIEP automatic suppression failure probabilities are generic industry values taken from the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide failure rates based on five different industry sources. The recommended generic values are: Water (3.8E-2 failure probability); Halon (5.9E-2 failure probability); and CO2 (4.0E-2). The NUREG-1150 automatic suppression system failure probabilities are generally consistent with the values provided in the EPRI FIVE Methodology, these are: Preaction and Deluge Systems (5.0E-2); Sprinkler Systems (2.0E-2); Halon (5.0E-2); and CO2 (4.0E-2).

6) The RMIEP manual suppression failure probabilities are based on LaSalle fire area specific analyses which consider: time to detection, time to assemble and suit-up, time to respond to scene, time to set-up at scene, time to search for fire source location, time to control fire. The RMIEP manual suppression analysis was supported by plant walkdowns, review of installed suppression system information, review of procedures and practices, and interviews with plant fire personnel.

7) Review of the RMIEP fire core damage cutsets and back-calculation of the CCDPs produces slightly (in the factor of 2-3 range) varying CCDPs for the same fire-induced damage states. This variance is due to cutset truncation limits and potential minor mis-interpretations of the fire-induced equipment damage (as represented in the RMIEP cutsets). Provided here for information.

Notes to Table C.4-2 (cont'd)

8) Per RMIEP, a small floor fire does not damage the cables of interest in this area.

9) Per RMIEP, a small floor fire is sufficient by itself to damage the cables of interest in this area (a large floor fire will also damage the cables of interest). However, the time to damage in either case is very similar and very quick (1-3 min.) for this small room (4E4-1), and the fire location area to room area ratio is the same in both the small and large fire scenarios (i.e., 1.0 - a small or large fire anywhere in the room is sufficient enough to damage the cables of interest), that RMIEP quantified an accident sequence for a single scenario (the small fire) rather than two scenarios. No large fire: small fire ratio was applied in the RMIEP frequency analysis for this fire area.

10) Per RMIEP, a large floor fire does not damage the cables of interest; however, due to the important cabling in the area, RMIEP assumes a very large fire (with a severity factor assumed to be half that of a large fire).

11) RMIEP documentation and/or quantification appears to be in error (although, the 4E4-1 fire scenario CDF is not significantly impacted given the similarity in train A and train B system importances). The documentation in Appendix B of the RMIEP fire analysis (NUREG/CR-4832, Vol. 9) states the following regarding equipment in fire location 4E4-1: "No equipment important to safety in this room. Train B cable spreading area." These two sentences appear conflicting; however, the quantification of this fire area, as documented on pp. F-51 thru F-56 of the RMIEP fire analysis, is an additional contradiction in that random failures of train B equipment are credited and train A equipment appears to be failed by the fire.

12) The RMIEP fire analysis modeled the Control Room with the following fire scenario: Fire starts in a Control Room panel/cabinet (1.85E-3/yr frequency), the fire is not suppressed before smoke requires abandonment of the Control Room (0.10 probability), and the operators do not successfully recover the plant from the Remote Shutdown Panel (6.4E-2 probability). However, The RMIEP fire analysis included a conservative evaluation of the Control Room fire frequency leading to core damage (1.39E-5/yr). Recent Exelon evaluations of control room fires indicates these conservative analyses are approximately a factor of ten too high in their CDF impact. Using this more realistic evaluation for the control room fires of the fire CDF results in a CDF for fires in the control room of 1.39E-6/yr.

13) The RMIEP switchgear cubicle fire is assigned a probability of 0.01 that the fire exits the top of the switchgear due to an inadequate seal; no area or severity ratios are applied.

REFERENCES

[C-1] ERIN Engineering and Research, Inc., Technical Evaluation of Extending LaSalle Diesel Generator Completion Time (CT) Using Probabilistic Risk Assessment Models for LaSalle, ERIN doc. #C1349911-4029, Rev. 2, March 2000.

Appendix B
LASALLE LERF CET EXTENSION

This appendix discusses modification of the LaSalle Revision 2003A Level 2 PSA LERF models for the purposes of this ILRT risk assessment to obtain additional release categories.

The LaSalle Level 2 PSA containment event tree structure and supporting documentation and analysis are based on the NRC specified requirements in RG 1.174 [B-14] to calculate a Large Early Release Frequency (LERF). The LaSalle Level 2 PSA provides the necessary information in risk-informed application submittals to the NRC as defined by RG 1.174. However, in seeking an exemption to the Integrated Leak Rate Test (ILRT) interval requirements, the NRC staff has requested additional information beyond the LERF estimate. This information includes the frequency of intact containment states along with radionuclide release effects for non-LERF end states. As this ILRT risk assessment requires evaluation of the full range of release magnitudes and timings, the LaSalle LERF model is extended here to address other release categories.

B.1 SUPPLEMENTARY CET NODES

Although the LaSalle Level 2 addresses specifically the LERF risk measure, the model structure and the Level 2 documentation also allows information to be developed regarding other (less severe) types of contributors to radionuclide release when supplemental analyses are performed.

The approach used to extend the LaSalle LERF Containment Event Tree (CET) models adds CET nodes to ask and resolve questions related to other critical safety functions that address the less severe (non-LERF) accident sequences. These supplementary CET nodes are added to the non-LERF accident sequences.

B.1.1 Radionuclide Release Categories

The radionuclide release category definitions are developed in the LaSalle Level 2 PSA documentation. The source term assignments are made using LaSalle specific calculations and BWR Mark II radionuclide release calculations from other industry studies.

The LaSalle Level 2 PSA uses the release severity and timing classification scheme described in Table B-1. The LaSalle LERF model of record is structured to explicitly track and quantify accident sequences resulting in the H/E (High magnitude Early release, i.e., LERF) release category.

B.1.2 Supplementary CET Nodes

The non-LERF accident sequences can be allocated to radionuclide release categories other than LERF (and including intact containment) through the development of supplementary CET nodes. These supplementary CET nodes can be quantified approximately based on the Level 1 cutsets, the previous failures in the CET, and the additional system and phenomenological effects associated with the supplemental nodes.

Figure B-1 shows the supplementary CET nodes that are considered appropriate for the allocation of non-LERF sequences. This CET development is based on numerous previous BWR Mark I and II containment CETs [B-1, 2, 3, 4, 5, 6, 7, 8, 9]. Table B-2 summarizes the definitions of these supplemental nodes.

The supplemental CET structure shown in Figure B-2 is sufficient to establish and answer the critical questions needed to distinguish among non-LERF radionuclide release end states. The quantification of the supplemental nodes (refer to Section B.2) and the assignment of release categories varies with the core damage accident class and CET sequence.

Table B-1

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME[1]

Release Severity		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Initial Release[2] Relative to Time for General Emergency Declaration
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Medium or Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 6 hours
Low-low (LL)	Less than 0.1		
No iodine (OK)	<<0.1		

(1) The combinations of severity and timing classifications results in one OK release category and 12 other release categories of varying times and magnitudes.

(2) The accident initiation is used as the surrogate for the time when EALs are exceeded.

Time of Release	Magnitude of Release			
	H	M	L	LL
E	H/E[1] (LERF)	M/E	L/E	LL/E
I	H/I	M/I	L/I	LL/I
L	H/L	M/L	L/L	LL/L

(1) LERF is equated to H/E – "high" magnitude of radionuclide release at an "early" time.

Figure B-1 SUPPLEMENTARY CET NODES

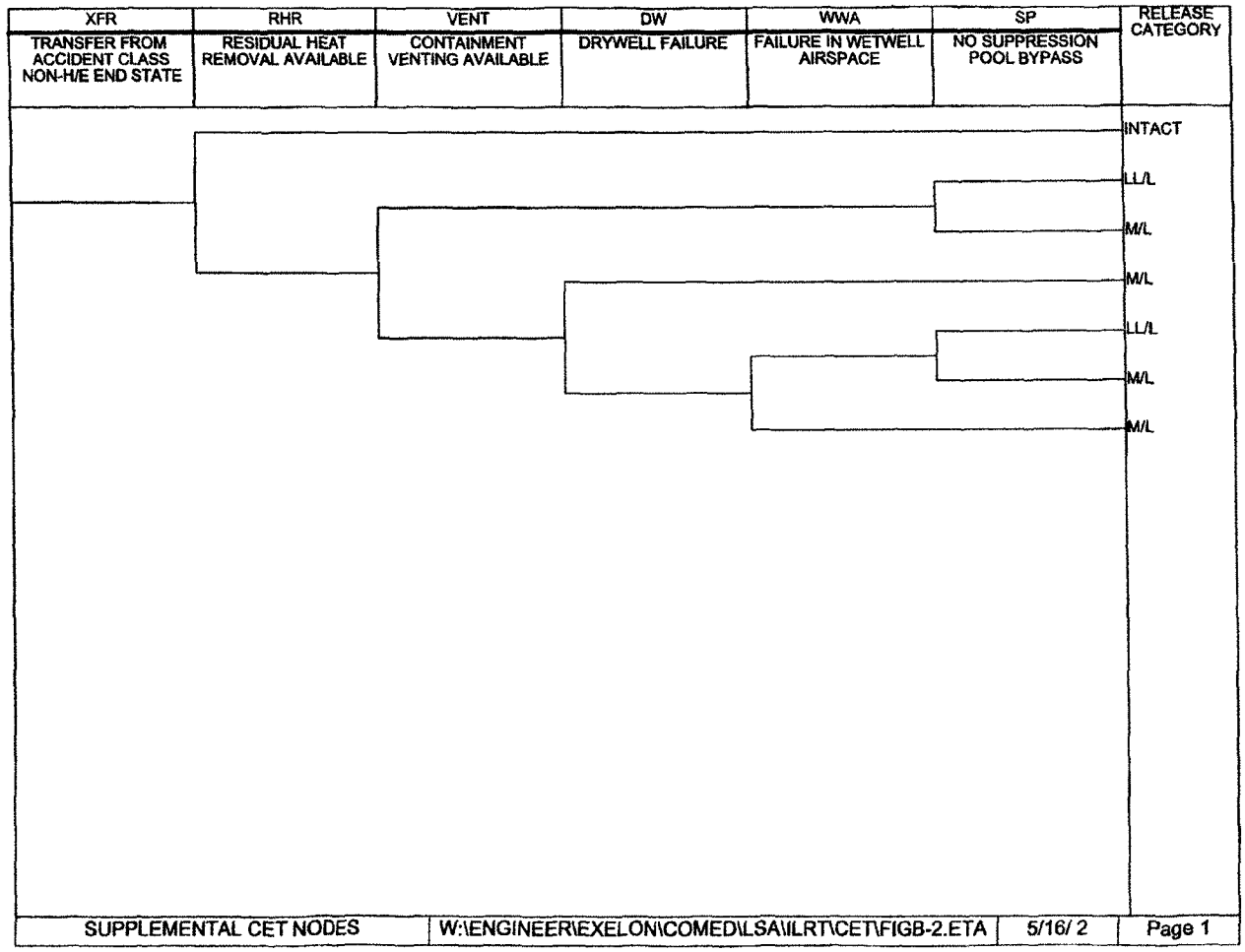


Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
RHR	This node addresses the availability of the RHR system and the operator action to initiate the system for containment heat removal. The RHR system, operating in suppression pool cooling mode, can maintain long term containment integrity through adequate containment heat removal if other failure modes can also be mitigated. The upward branch at this node represents successful containment heat removal via the RHR system operating in the suppression pool cooling mode. Sequences with successful suppression pool cooling lead to an endstate with an intact containment. The downward branch models failure of RHR suppression pool cooling. Sequences with unsuccessful suppression pool cooling will lead to some containment release, either through use of the EOP-directed containment vent or through a containment breach caused by over-temperature and pressure failure.
VENT	This node models use of the wetwell vent to relieve containment pressure in the event of RHR suppression pool cooling failure. Containment venting provides the operator a means of removing decay heat and non-condensible gases, and maintaining containment integrity. The upward branch at this node represents successful use of the containment vent, and release of fission products. Subsequent node SP will determine whether or not the release of fission products is scrubbed by the suppression pool water. The downward branch at this node represents failure of the containment vent. Failure of RHR and VENT will eventually result in containment failure and release of fission products. Subsequent nodes will question whether the containment failure occurs in the drywell or the wetwell, and whether the release is scrubbed by the suppression pool water.
DW	The upward branch of this node indicates containment failure occurs in the drywell. Releases are characterized assuming the drywell failure is at the Drywell head and are in the Moderate magnitude range. The timing of the release is Late given the lengthy time required to overpressurize the primary containment. The downward branch of this node indicates containment failure occurs in the wetwell. Subsequent nodes question whether the wetwell failure occurs in the wetwell airspace or below the waterline, and whether the release is scrubbed by the suppression pool water.

Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
WWA	If the containment failure does not occur in the drywell then it occurs in the wetwell, either in the wetwell airspace region or below the wetwell waterline. The upward branch of this node indicates containment failure occurs in the wetwell airspace. The subsequent SP node questions whether the radionuclide releases are scrubbed or not. The downward branch of this node indicates containment failure occurs below the wetwell waterline. The model assumes that the wetwell failure location is such that the containment breach is not submerged by the pool level. As such, the release associated with this pathway are similar to that of a drywell release.
SP	This node models potential bypass of the containment vapor suppression system (VSS) to determine whether or not releases through the containment vent or via a breach in the wetwell are scrubbed by the pool water. The vapor suppression system (VSS) is composed of the suppression pool, vent pipes, internal ring header, downcomers that connect the drywell to the torus, discharge lines from the relief valves to the suppression pool, the vacuum breakers between the wetwell and the drywell, and the overall boundary between the drywell and the wetwell. The principal function of the VSS is to control containment pressure by condensing steam. In severe accidents in which core damage has occurred, the system also directs potential radionuclide releases to be scrubbed in the suppression pool. The scrubbing of fission products in the suppression pool represents a significant removal mechanism for fission products. The suppression pool can act as an effective scrubber of fission products when it is maintained in the path of radionuclide releases. Possible ways that the suppression pool can be bypassed, and therefore, scrubbing effectiveness diminished, is if: (1) a breach is created between the drywell and the wetwell; (2) wetwell to drywell vacuum breakers fail open; or (3) suppression pool water level decreases below the bottom of the downcomers. If loss of the vapor suppression function (i.e., suppression pool bypass) occurs after the molten core has penetrated the reactor vessel, the effectiveness of continued fission product scrubbing could be compromised. This CET heading is used to estimate the split fraction related to suppression pool bypass; and therefore, to characterize the magnitude of radionuclides that may escape the containment if wetwell

Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
	failure or venting occurs.
SP (con't)	The downward branch of this node indicates that radionuclides bypass the suppression pool water due to one or more of the following failures: • Wetwell to drywell vacuum breaker stuck open • Suppression pool water level below the bottom of the downcomers • Vent pipes or downcomers breached during the core melt progression Releases associated with this pathway are similar to that of a drywell release. The upward branch of this node indicates that radionuclides are directed through the suppression pool (i.e., no suppression pool bypass), this requires that none of the above failures occurs. The magnitude of scrubbed releases is two magnitude classifications lower than that of unscrubbed releases.

These supplemental CET nodes are added to the non-LERF sequences of the "no initial containment failure" accident classes (i.e., Class I's, IIIB, and IIIC).

The supplemental CET nodes are not added to accidents in which the containment has already failed (i.e, Classes II, IIID, IV, and V). Sufficient information exists in the LERF CETs for these accident classes to enable assignment of release categories for the non-LERF sequences.

B.2 SUPPLEMENTARY CET NODAL QUANTIFICATION

The LaSalle Level 1 cutset results by accident class were reviewed to identify the dominant contributors to each accident class. Based on these cutsets, the supplemental CET nodes are quantified on a conditional basis. These conditional failure probabilities reflect the functional and support system failures that have occurred in the Level 1 PSA analysis and prior CET nodes. These conditional failure probabilities reflect the dependencies from the Level 1 cutsets and also account for degraded plant conditions and operating environment.

Table B-4 summarizes the quantification of the failure probabilities for the supplemental CET nodes.

B.2 RESULTS OF EXTENDED CETS

The quantified LaSalle extended CETs are provided in Attachment B-1. The results are summarized in Table B-7.

Table B-4

SUMMARY OF SUPPLEMENTAL NODAL QUANTIFICATION (DOWN BRANCHES)

Node ID	Quantification
RHR	The base RHR suppression pool cooling (SPC) failure probability with support systems intact is approximately 1E-3 (based on Level 1 PSA model gate SPC). The failure probability for a single train of RHR SPC is approximately 2E-2 (based on Level 1 PSA model gate RHR-TRAINA-SP). These two failure probabilities are used in most cases for the RHR node. Refer to Table B-5 for a detailed summary of the RHR conditional failure probabilities used in each supplemental CET.
VENT	The conditional failure probability of containment heat removal via venting is dependent on the availability of DC power and Instrument Air. The conditional failure probability of containment venting is negligibly impacted by previous failure of the RHR system. The failure probability for containment venting given SPC failure is approximately 4E-2 (based on Level 1 PSA model gates PCV and SPC). Estimation of the VENT conditional failure probability is based on review on the Level 1 cutsets. In all cases, the conditional failure probability of 1E-1 is used. The 1E-1 value is used instead of the base 4E-2 value to account for the potential increase in the containment venting HEP during post-core damage accident scenarios. Refer to Table B-6 for a detailed summary of the basis for the 1E-1 failure probability for each supplemental CET.
DW	The downward branch of the DW supplemental CET node indicates containment failure occurs in the wetwell. Based on the containment structural evaluation of the Level 2 PSA, the probability of failure in the wetwell (and not in the DW) is 2.47E-1 (0.1172 + 0.1111 + 0.0183) for accident Classes I and III given core melt progression, no containment heat removal but TD = S. (See Table 3.2-3 of the LaSalle Level 2 PSA.).
WWA	The downward branch of the WWA supplemental CET node indicates containment failure below the wetwell waterline. Based on the containment structural evaluation of the LaSalle Level 2 PSA, the conditional probability of failure in the wetwell waterspace (and not the wetwell airspace) is 7.42E-2 (0.0183/(0.1172+0.1111+0.0183)) for accident Classes I and III given core melt progression, no containment heat removal but TD = S. (See

Table B-4

SUMMARY OF SUPPLEMENTAL NODAL QUANTIFICATION (DOWN BRANCHES)

Node ID	Quantification
	Table 3.2-3 of the Level 2 PSA.).
SP	The following three suppression pool bypass conditional failure probabilities are used: • 2.1E-3 • 2.1E-2 • 1.0 The 2.1E-3 SP failure probability applies to non-LOCA scenarios in which core melt is successfully arrested in-vessel. This failure mode is derived from NRC modeling of fission product transport in the MARCH code in which Sandia postulated a potential bypass mechanism which can occur early in a scenario resulting in high concentration of volatile fission products in the wetwell airspace, and subsequent suppression pool bypass (dominated by the coincidental random failures of SRV discharge vacuum breakers and WW-DW vacuum breakers.) The 2.1E-2 SP failure probability applies to LOCA sequences where steam is discharged directly to the drywell, but where no core debris is discharged to the drywell. The 1.0 SP failure probability applies to scenarios in which the RPV is breached by the core damage progression (these scenarios are addressed in the Page 2 supplemental CETs). As discussed in Section C.6 of the LaSalle Level 2 PSA, the drywell sumps are adequate to hold approximately 30% of the core debris; however, it is estimated that eventually more than 80% of the core debris may be released from the RPV causing the sumps to overflow. The overflowing core debris is postulated to contact and fail (in under an hour following RPV breach) the drywell downcomers, thus leading to suppression pool bypass.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IA	• RHR not asked in IA Level 1 accident sequences • Approximately 20% of IA cutsets involve loss of one DC division	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Although RHR is not asked in the Level 1, a significant percentage of Class IA cutsets involve loss of a division of DC. Therefore, it is reasonably assumed that only 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.
IBE	• RHR not asked in IBE Level 1 accident sequences • No AC power available in IBE Level 1 scenarios	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.
IBL	• RHR not asked in IBL Level 1 accident sequences • No AC power available in IBL Level 1 scenarios	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IC	• RHR asked in some IC Level 1 accident sequences • IC cutsets dominated by operator failure to emergency depressurize and not by LP injection equipment failure	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Although some Class IC sequences ask RHR, the majority of Class IC cutsets are due to operator failure to perform RPV emergency depressurization. This nodal probability assumes that at least 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.
ID	• RHR asked in ID Level 1 accident sequences • LP ECCS failures present in most, if not all, ID cutsets	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	0.5	RHR has been asked and has failed in the Level 1 Class ID sequences. Although an injection source has been recovered in the Level 2, this nodal probability assumes that the recovered system may not be an RHR train.
IE	• RHR asked in IE Level 1 accident sequences • 100% of IE cutsets involve failure of both divisions of DC	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Recovery of injection in the Level 2 is most likely due to recovery of one division of DC power. Therefore, it is reasonably assumed that only 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IIIB	• RHR not asked in IIIB Level 1 accident sequences • IIIB cutset dominated by operator failure to emergency depressurize	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	RHR is not asked in the Level 1 and the Class IIIB cutsets are not dominated by support system failures. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.
IIIC	• RHR asked in IIIC Level 1 accident sequences • LP ECCS failures present in most, if not all, IIIC cutsets	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	0.5	RHR has been asked and has failed in the Level 1 Class IIIC sequences. Although an injection source has been recovered in the Level 2, this nodal probability assumes that the recovered system may not be an RHR train.

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IA	• Vent not asked in IA Level 1 accident sequences • Approximately 20% of IA cutsets involve loss of one DC division	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IBE	• Vent not asked in IBE Level 1 accident sequences • No AC power available in IBE Level 1 scenarios	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IBL	• Vent not asked in IBL Level 1 accident sequences • No AC power available in IBL Level 1 scenarios	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IC	• Vent asked in some IC Level 1 accident sequences • IC cutsets dominated by operator failure to emergency depressurize and not by LP injection equipment failure	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. The majority of Class IC cutsets are due to operator failure to emergency depressurize the RPV. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
ID	• Vent asked in ID Level 1 accident sequences • LP ECCS failures present in most, if not all, cutsets	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. A minor percentage of Class ID cutsets contain AC or IA failures that would impact VENT. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IE	• Vent asked in IE Level 1 accident sequences • 100% of IE cutsets involve failure of both divisions of DC	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	100% of the Class IE cutsets are loss of DC events; divisional DC failures have no impact on the VENT failure probability. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IIIB	• Vent not asked in IIIB Level 1 accident sequences • Cutset dominated by operator failure to emergency depressurize	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Class IIIB cutsets are not dominated by support system failures. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IIIC	• Vent asked in IIIC Level 1 accident sequences • LP ECCS failures present in most, if not all, cutsets	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. A minor percentage (~10%) of Class IIIB cutsets contain AC or IA failures that would impact VENT. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-7

SUMMARY OF LASALLE UNIT 2 LEVEL 2 PSA RESULTS

Level 1 CDF		LaSalle Level 2 PSA Release Bin Frequencies[1], [2]													
Class	NEW CDF	Intact (OK)	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E[5]	H/I	H/L	Total Release
IA/IE	2.42E-07	2.20E-07	0.00E+00	0.00E+00	3.77E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.47E-08	7.10E-10	2.07E-09	0.00E+00	0.00E+00	2.12E-08
IBE	5.67E-07	3.71E-07	0.00E+00	0.00E+00	1.92E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.86E-07	1.81E-10	6.51E-09	0.00E+00	0.00E+00	1.93E-07
IBL	7.13E-07	3.97E-07	0.00E+00	0.00E+00	1.82E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.06E-07	2.15E-10	0.00E+00	1.06E-08	0.00E+00	3.17E-07
IC	6.41E-09	6.20E-09	0.00E+00	0.00E+00	1.17E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.99E-12	1.01E-11	2.34E-11	0.00E+00	0.00E+00	1.52E-10
ID	1.42E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.40E-06	0.00E+00	2.18E-08	0.00E+00	0.00E+00	1.42E-06
II[3]	3.65E-06	0.00E+00	0.00E+00	2.68E-07	0.00E+00	0.00E+00	2.40E-06	0.00E+00	0.00E+00	8.81E-08	0.00E+00	1.40E-11	8.93E-07	0.00E+00	3.65E-06
IIIA	1.00E-09	5.69E-10	0.00E+00	0.00E+00	7.72E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.03E-10	3.89E-12	2.89E-08	0.00E+00	0.00E+00	2.93E-08
IIIB	9.39E-09	9.26E-09	0.00E+00	0.00E+00	8.38E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.29E-12	8.96E-13	7.42E-11	0.00E+00	0.00E+00	8.48E-11
IIIC	5.70E-08	1.74E-08	0.00E+00	0.00E+00	8.15E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.14E-08	9.22E-09	8.45E-10	0.00E+00	0.00E+00	3.96E-08
IIID	7.29E-08	-3.30E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.29E-08	0.00E+00	0.00E+00	7.29E-08
IV[4]	1.61E-07	0.00E+00	1.09E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.79E-08	0.00E+00	0.00E+00	5.13E-08	0.00E+00	0.00E+00	1.30E-07
V	1.71E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.71E-07	0.00E+00	0.00E+00	1.71E-07
Total	7.07E-06[6]	1.02E-06	1.09E-08	2.68E-07	1.24E-08	0.00E+00	2.40E-06	0.00E+00	6.79E-08	2.01E-06	1.03E-08	3.55E-07	9.03E-07	0.00E+00	6.04E-06
% of Total CDF:		14.4%	0.2%	3.8%	0.2%	0.0%	33.9%	0.0%	1.0%	28.5%	0.1%	5.0%	12.8%	0.0%	85.5%
% of Total Release:		N/A	0.2%	4.4%	0.2%	0.0%	39.7%	0.0%	1.1%	33.3%	0.2%	5.9%	14.9%	0.0%	100.0%

Notes to Table B-7:

(1) Release bin nomenclature is [Release Magnitude]/[Timing of Release], where:

LL:	Low-Low	E:	Early
L:	Low	I:	Intermediate
M:	Moderate	L:	Late
H:	High		

(2) The LaSalle Revision 2001A Level 2 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences.

(3) Includes all Class II subcategories.

(4) Includes contributions from Class IVL.

(5) LERF values are calculated directly by the LERF CET from the 2003A PRA.

(6) The CDF used in the ILRT analysis is the result of the PRA Quant calculation used to develop the Accident Class results. This calculation has some non-minimal accident sequences and the resulting CDF is 7.07E-6/yr. This CDF is slightly conservative for the purposes of this analysis.

REFERENCES

[B-1] *Peach Bottom Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Philadelphia Electric Company, February 1992.

[B-2] *Nine Mile Point Unit 1 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Niagara Mohawk Power Corporation, March 1994.

[B-3] *Duane Arnold Energy Center Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Iowa Electric Light and Power Company, December 1992.

[B-4] *Cooper Nuclear Station Level 2 PRA,* NPPD, 1998.

[B-5] *Fermi 2 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Detroit Edison Company, September 1991.

[B-6] *Limerick Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Philadelphia Electric Company, March 1992.

[B-7] *Quad Cities* Level *2/LERF Evaluation,* ERIN Engineering and Research, Inc. for ComEd, July 1999.

[B-8] *Nine Mile Point Unit 2 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Niagara Mohawk Power Corporation, January 1992.

[B-9] *Brunswick Level 2/LERF Evaluation,* ERIN Report No. C1100001-4265, November 2000.

[B-10] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[B-11] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.

[B-12] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,* EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

[B-13] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

REFERENCES (cont'd)

[B-14] Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* July 1998.

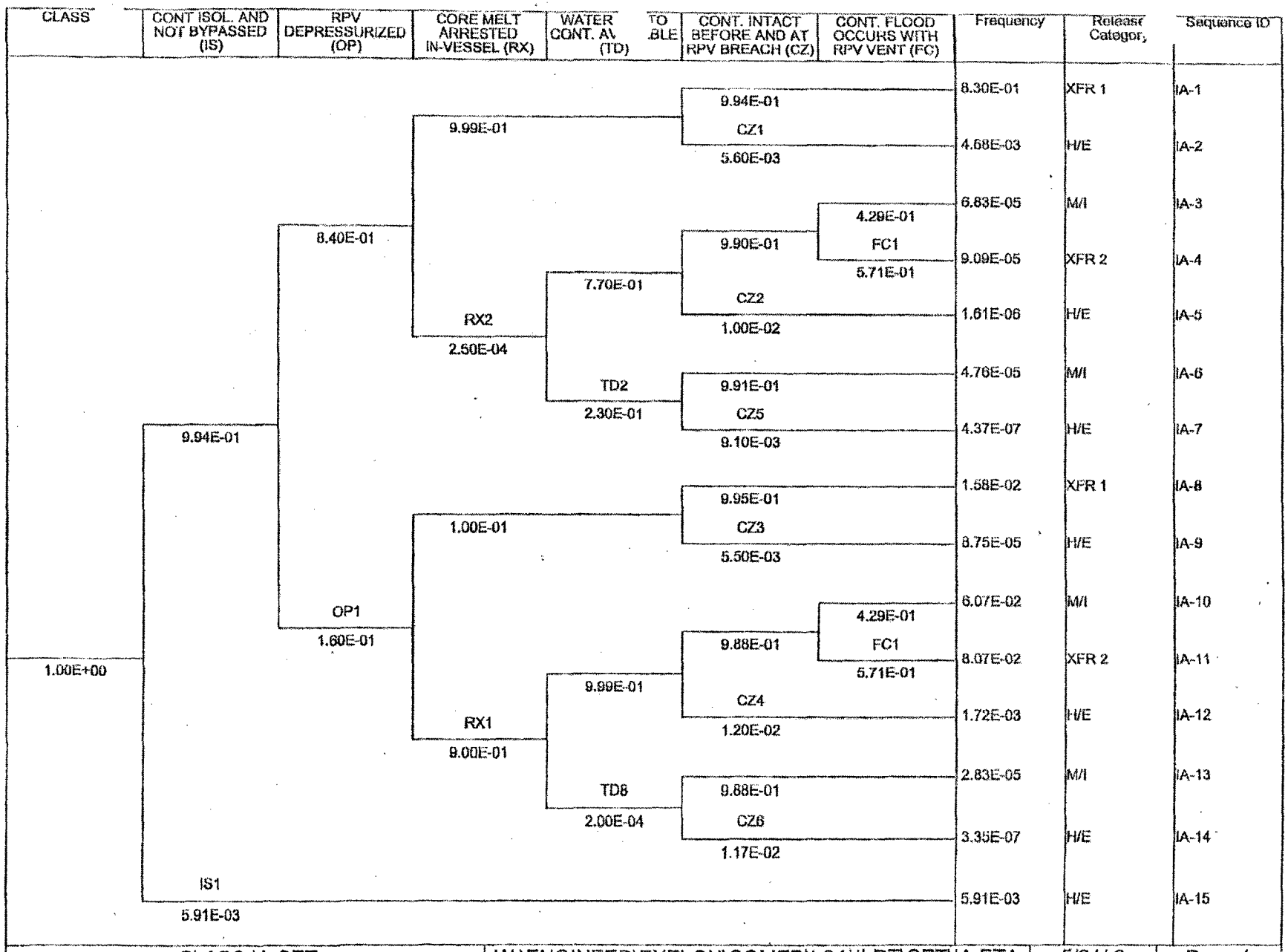
Attachment B1
QUANTIFIED LASALLE EXTENDED CETs

This attachment provides the quantified LaSalle extended containment event trees. The following quantified CETs are included in this attachment:

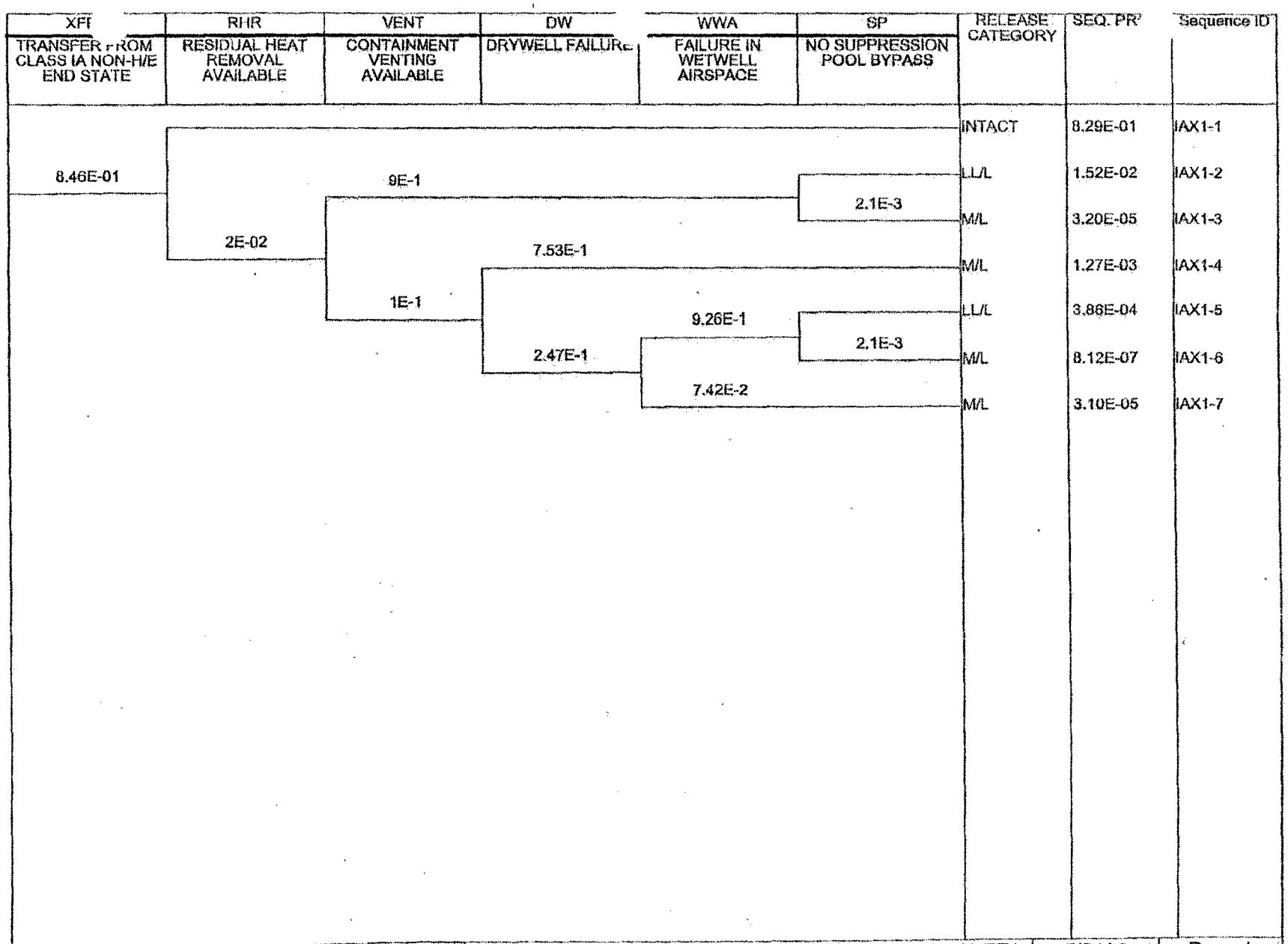
- Class IA CET
- Supplemental CET for Class IA (Page 1)
- Supplemental CET for Class IA (Page 2)
- Class IBE CET
- Supplemental CET for Class IBE (Page 1)
- Supplemental CET for Class IBE (Page 2)
- Class IBL CET
- Supplemental CET for Class IBL (Page 1)
- Supplemental CET for Class IBL (Page 2)
- Class IC CET
- Supplemental CET for Class IC (Page 1)
- Supplemental CET for Class IC (Page 2)
- Class ID CET
- Supplemental CET for Class ID (Page 1)
- Supplemental CET for Class ID (Page 2)
- Class IE CET
- Supplemental CET for Class IE (Page 1)
- Supplemental CET for Class IE (Page 2)
- Class II CET
- Class IIIB CET
- Supplemental CET for Class IIIB (Page 1)
- Supplemental CET for Class IIIB (Page 2)
- Class IIIC CET
- Supplemental CET for Class IIIC (Page 1)
- Supplemental CET for Class IIIC (Page 2)

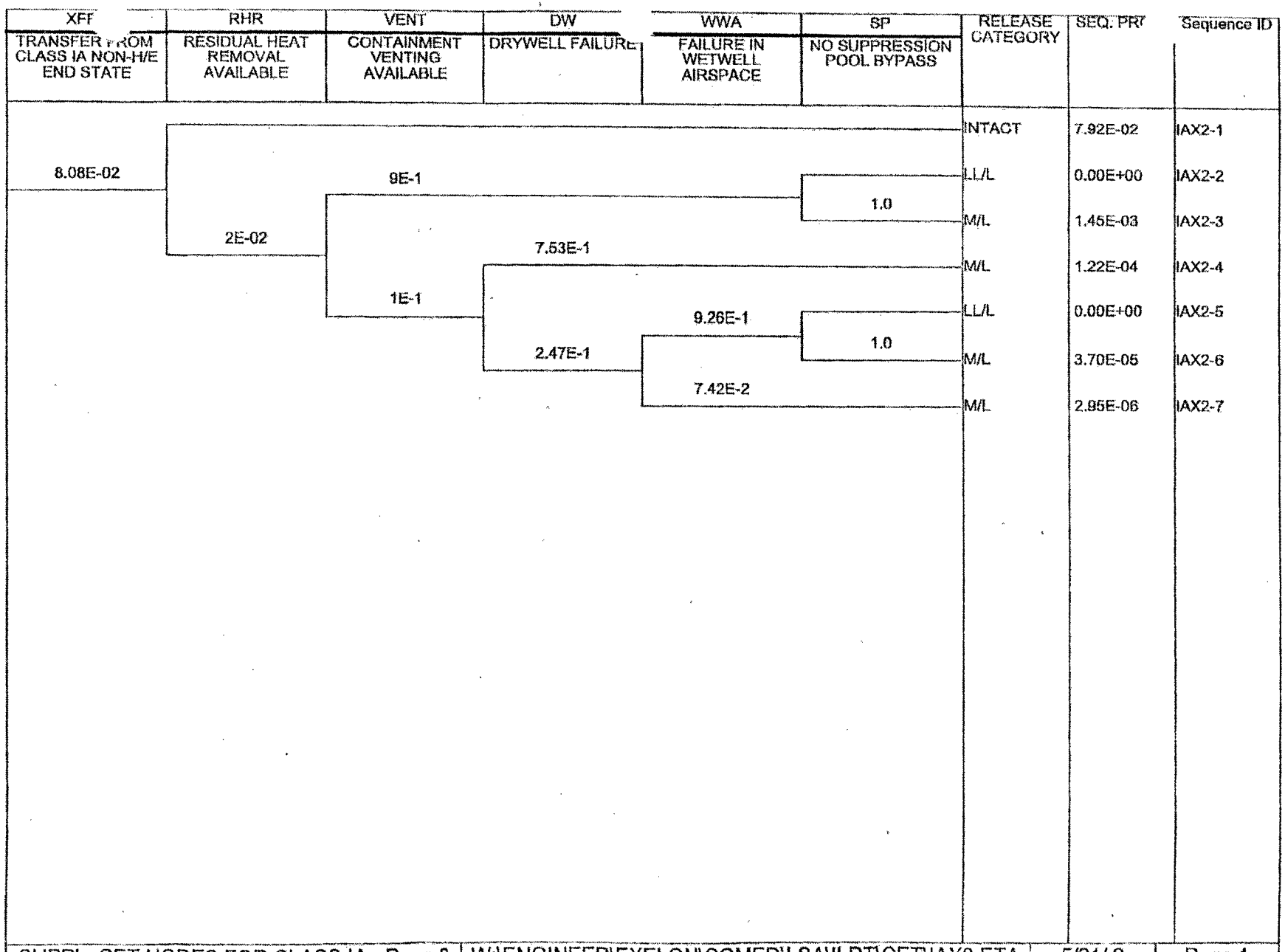
- Class IID CET
- Class IV CET
- Class V CET

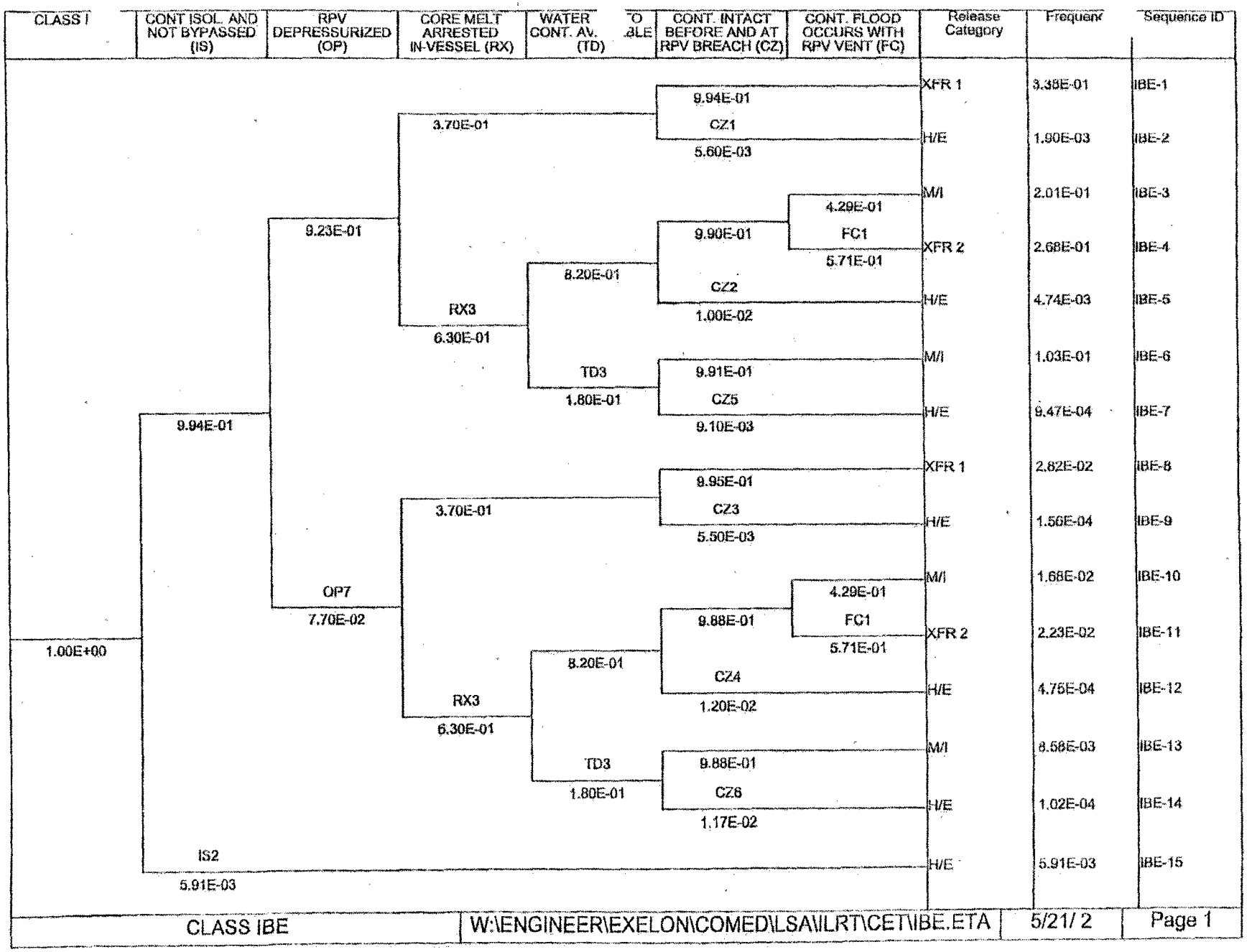
As the CETs use only point estimates (i.e., no cutsets or fault tree logic are input into these CETs), the CETs are developed and quantified using the ETA event tree code. As can be seen from the attached quantified CETs, the incoming accident class information for each CET is entered as a 1.00 point estimate. As such, the CETs calculate conditional release categories. The individual sequences are summed according to release category and the totals are then multiplied in a spreadsheet by the individual accident class subtotals to determine the release category frequencies. The results are summarized in Table B-7.



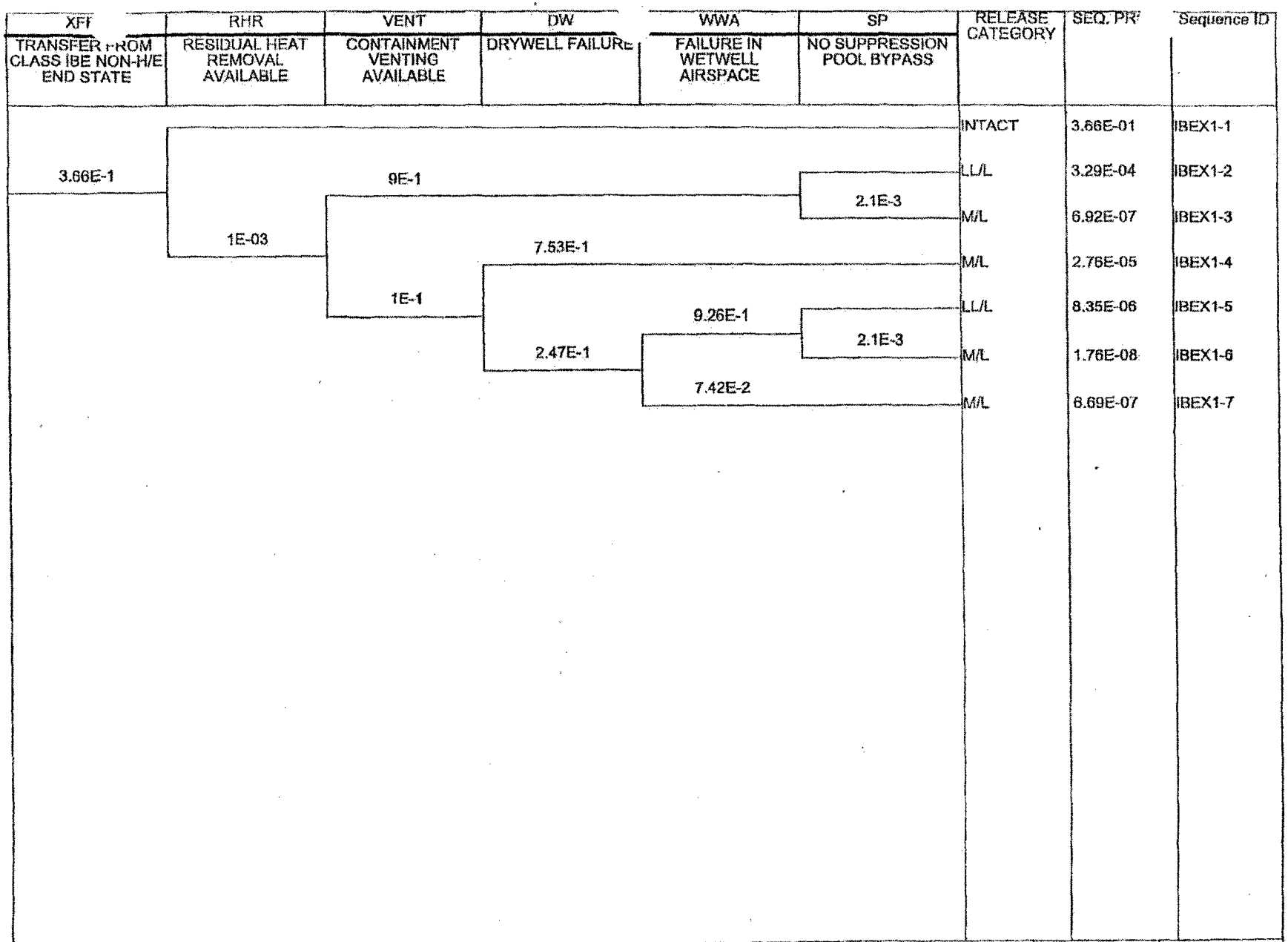
CLASS IA CET	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IA.ETA	5/21/ 2	Page 1

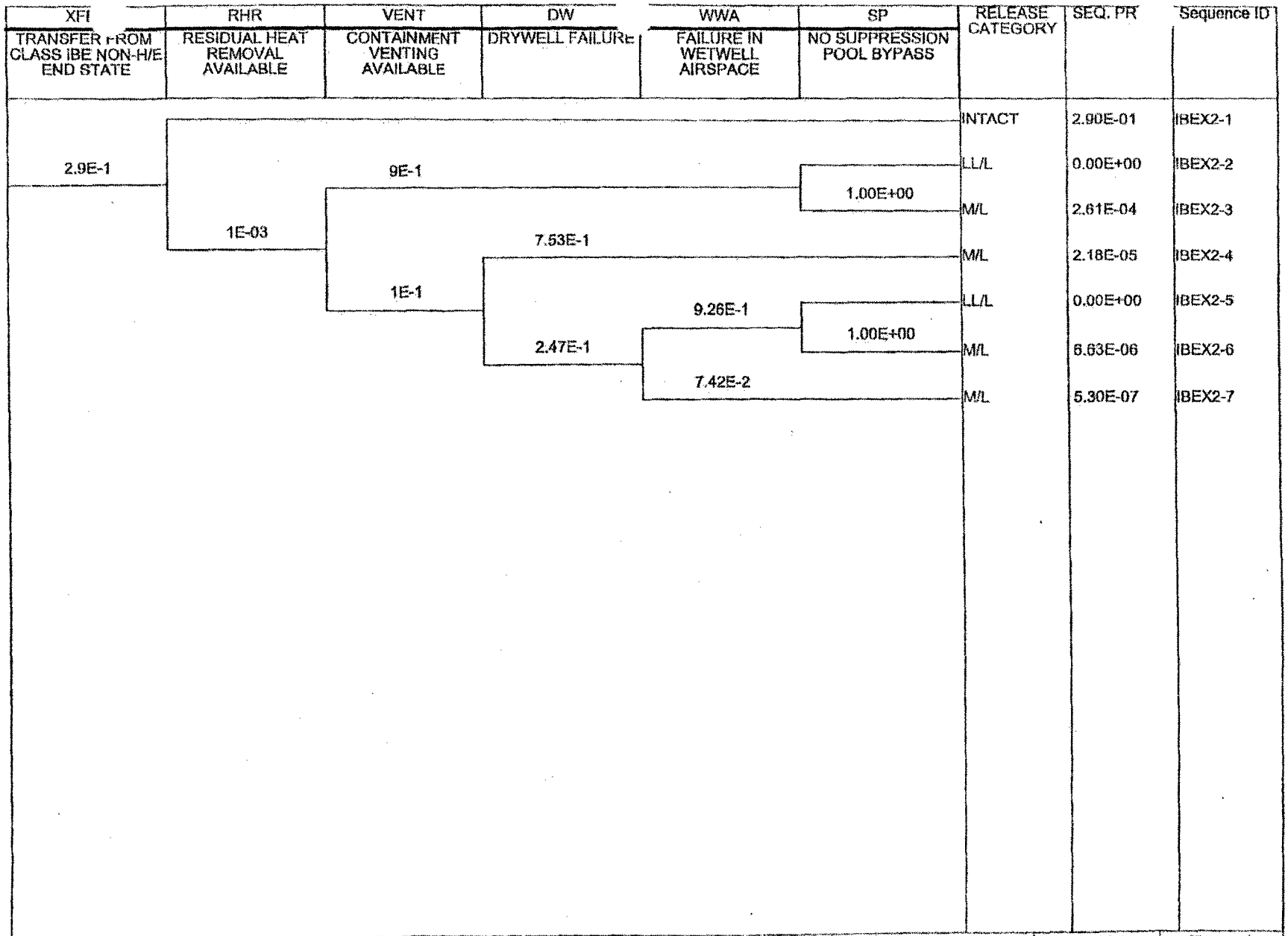

XFI
TRANSFER FROM CLASS IA NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
8.46E-01
2E-02
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-3
2.1E-3
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
8.29E-01
1.52E-02
3.20E-05
1.27E-03
3.86E-04
8.12E-07
3.10E-05
IAX1-1
IAX1-2
IAX1-3
IAX1-4
IAX1-5
IAX1-6
IAX1-7

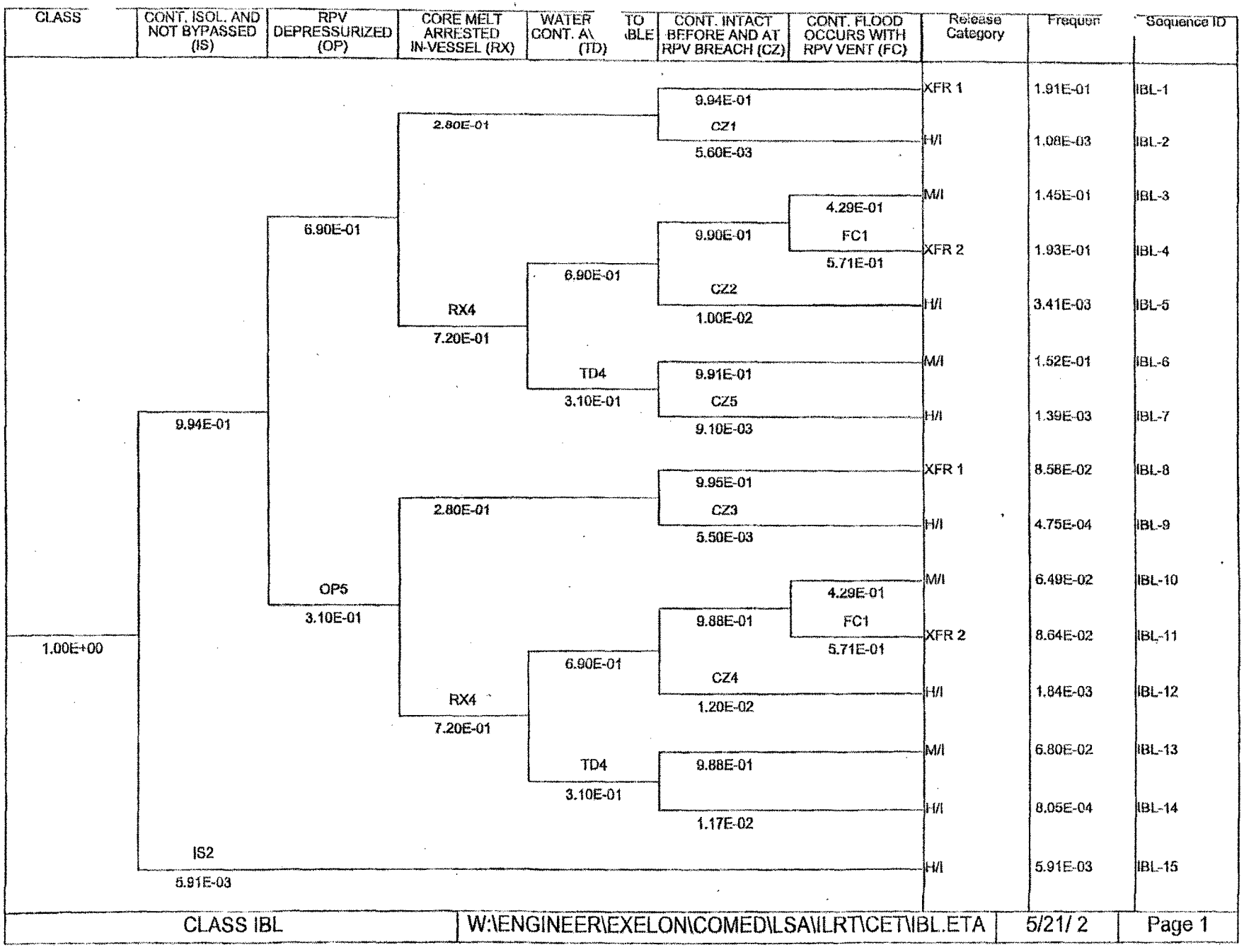

XFF
TRANSFER FROM CLASS IA NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
8.08E-02
2E-02
9E-1
1E-1
1.0
7.53E-1
9.26E-1
2.47E-1
1.0
7.42E-2
INTACT 7.92E-02 IAX2-1
LL/L 0.00E+00 IAX2-2
M/L 1.45E-03 IAX2-3
M/L 1.22E-04 IAX2-4
LL/L 0.00E+00 IAX2-5
M/L 3.70E-05 IAX2-6
M/L 2.95E-06 IAX2-7



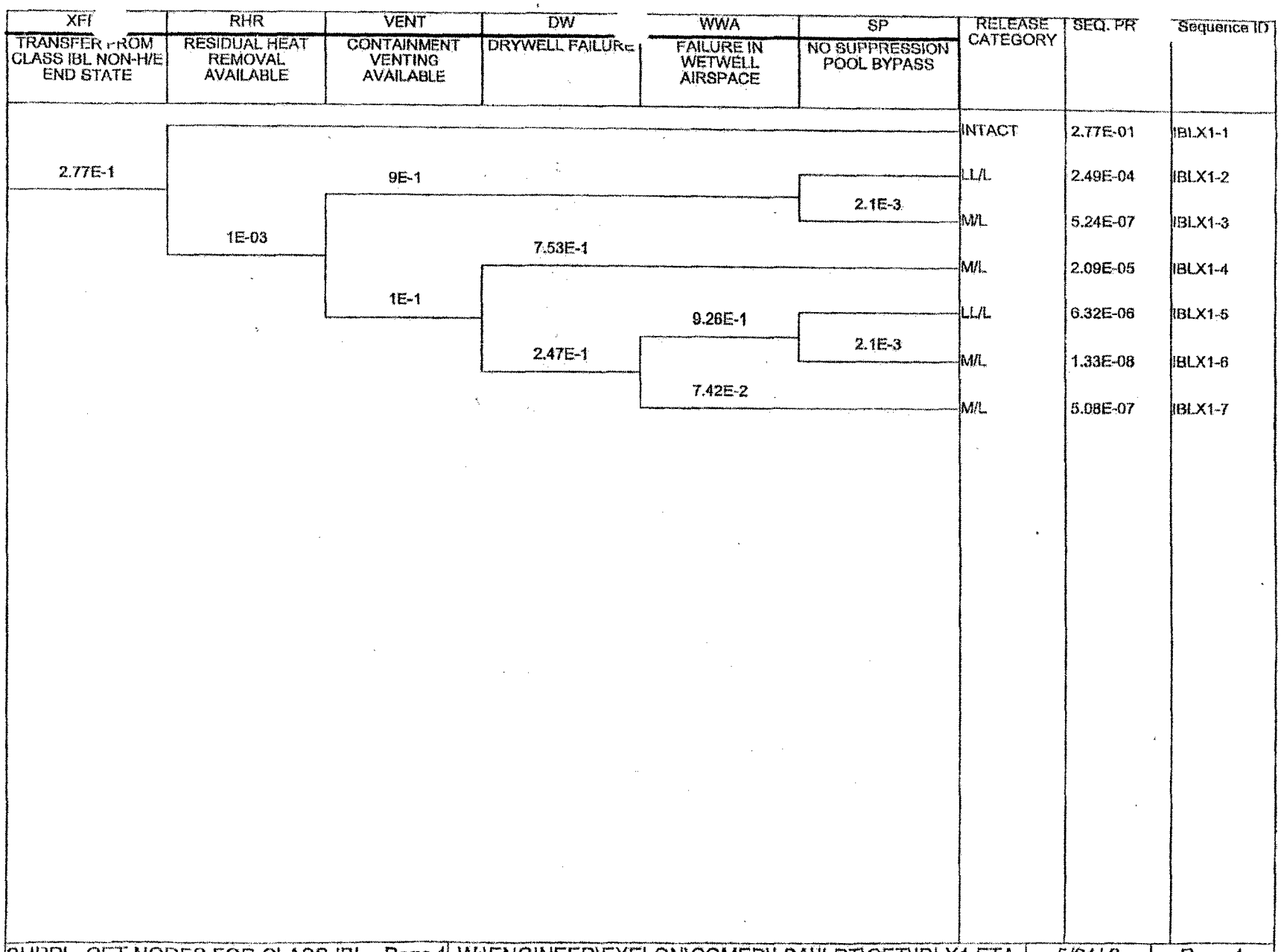
CLASS IBE | W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IBE.ETA | 5/21/ 2 | Page 1

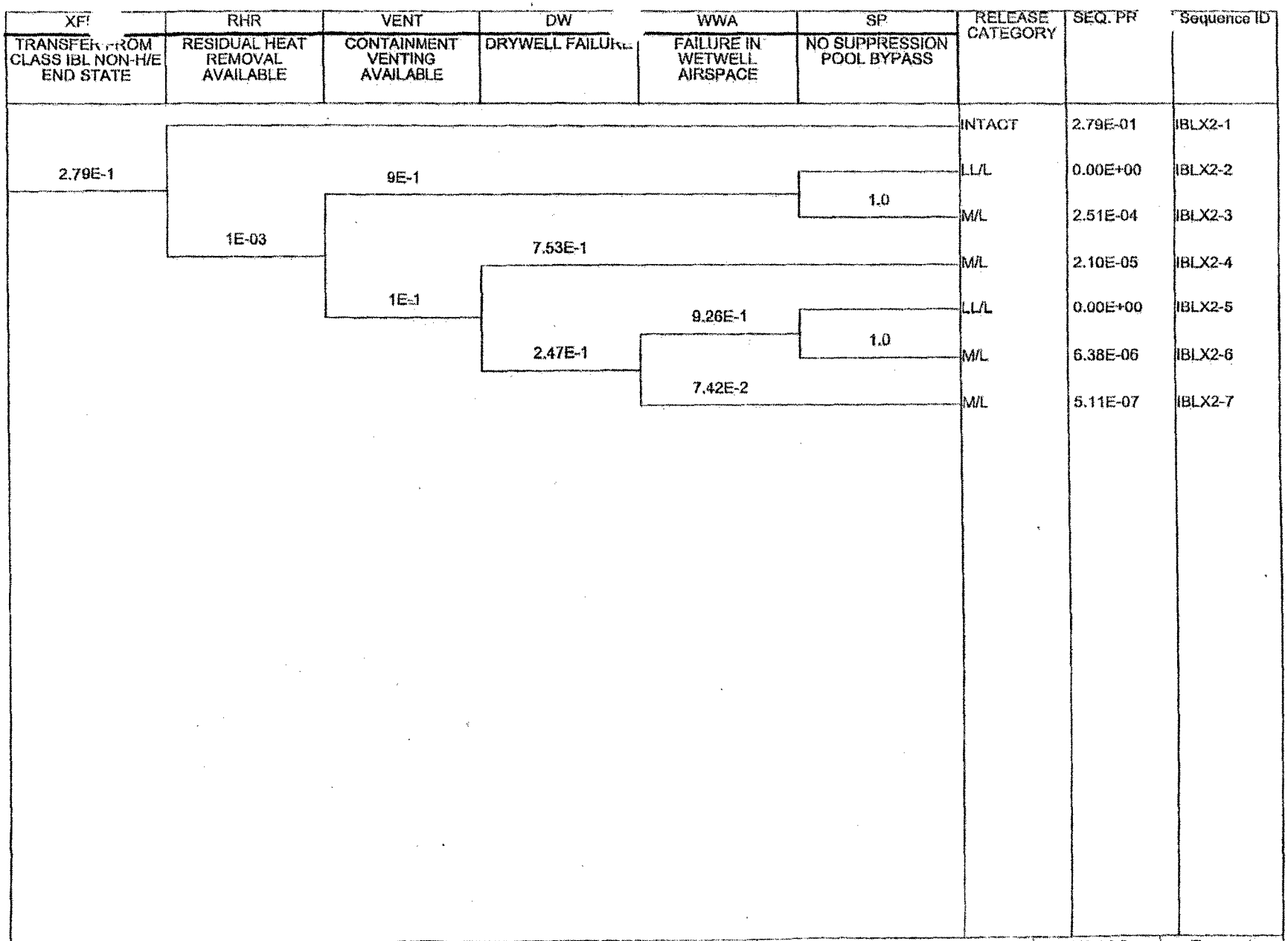

XFF
TRANSFER FROM CLASS IBE NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
3.66E-1
1E-03
9E-1
2.1E-3
7.53E-1
1E-1
9.26E-1
2.47E-1
2.1E-3
7.42E-2
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
3.66E-01
3.29E-04
6.92E-07
2.76E-05
8.35E-06
1.76E-08
6.69E-07
IBEX1-1
IBEX1-2
IBEX1-3
IBEX1-4
IBEX1-5
IBEX1-6
IBEX1-7

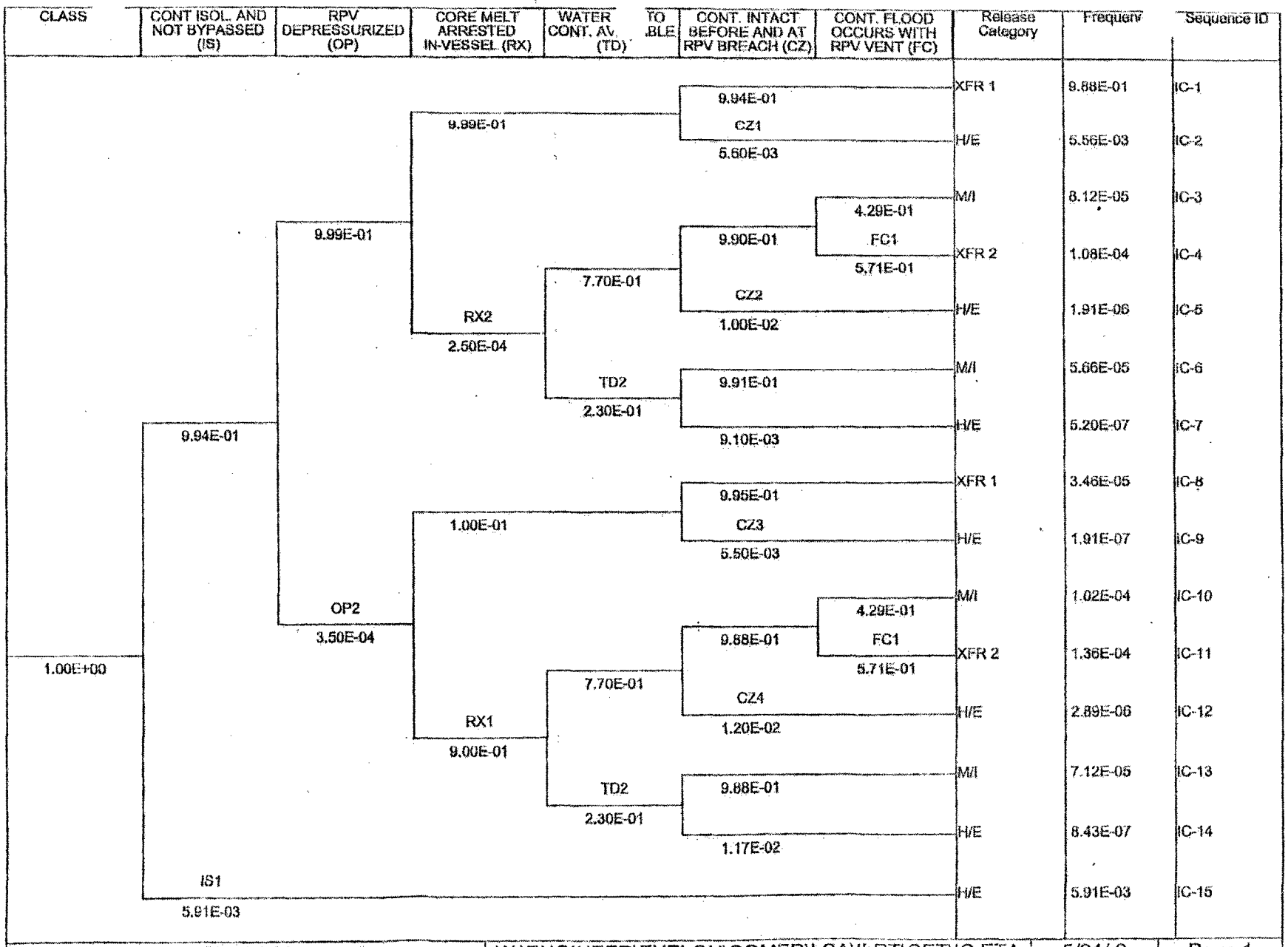

XFI
TRANSFER FROM CLASS IBE NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
2.9E-1
1E-03
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
1.00E+00
1.00E+00
INTACT 2.90E-01 IBEX2-1
LL/L 0.00E+00 IBEX2-2
M/L 2.61E-04 IBEX2-3
M/L 2.18E-05 IBEX2-4
LL/L 0.00E+00 IBEX2-5
M/L 6.63E-06 IBEX2-6
M/L 5.30E-07 IBEX2-7



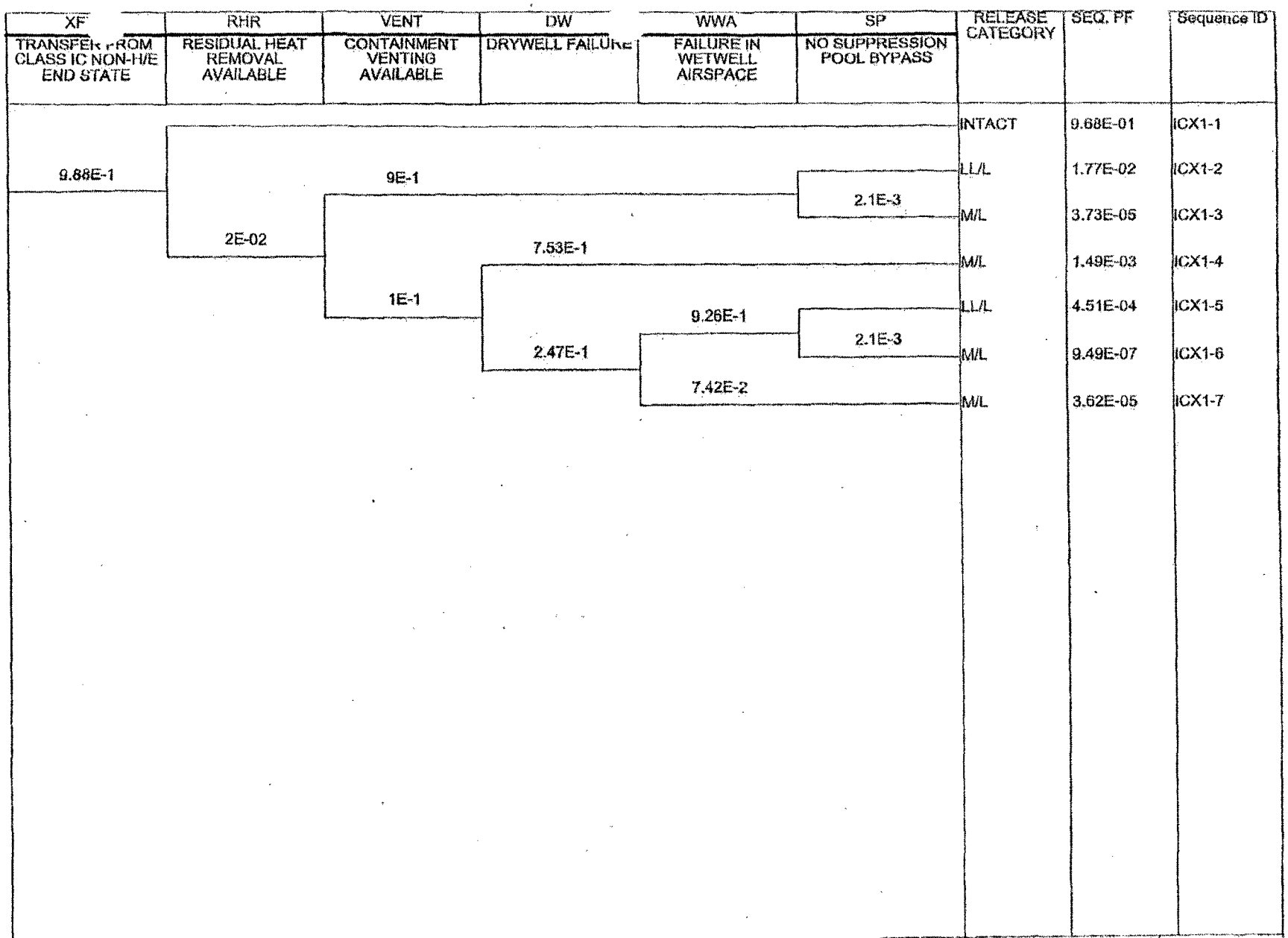
CLASS IBL	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IBL.ETA	5/21/ 2	Page 1

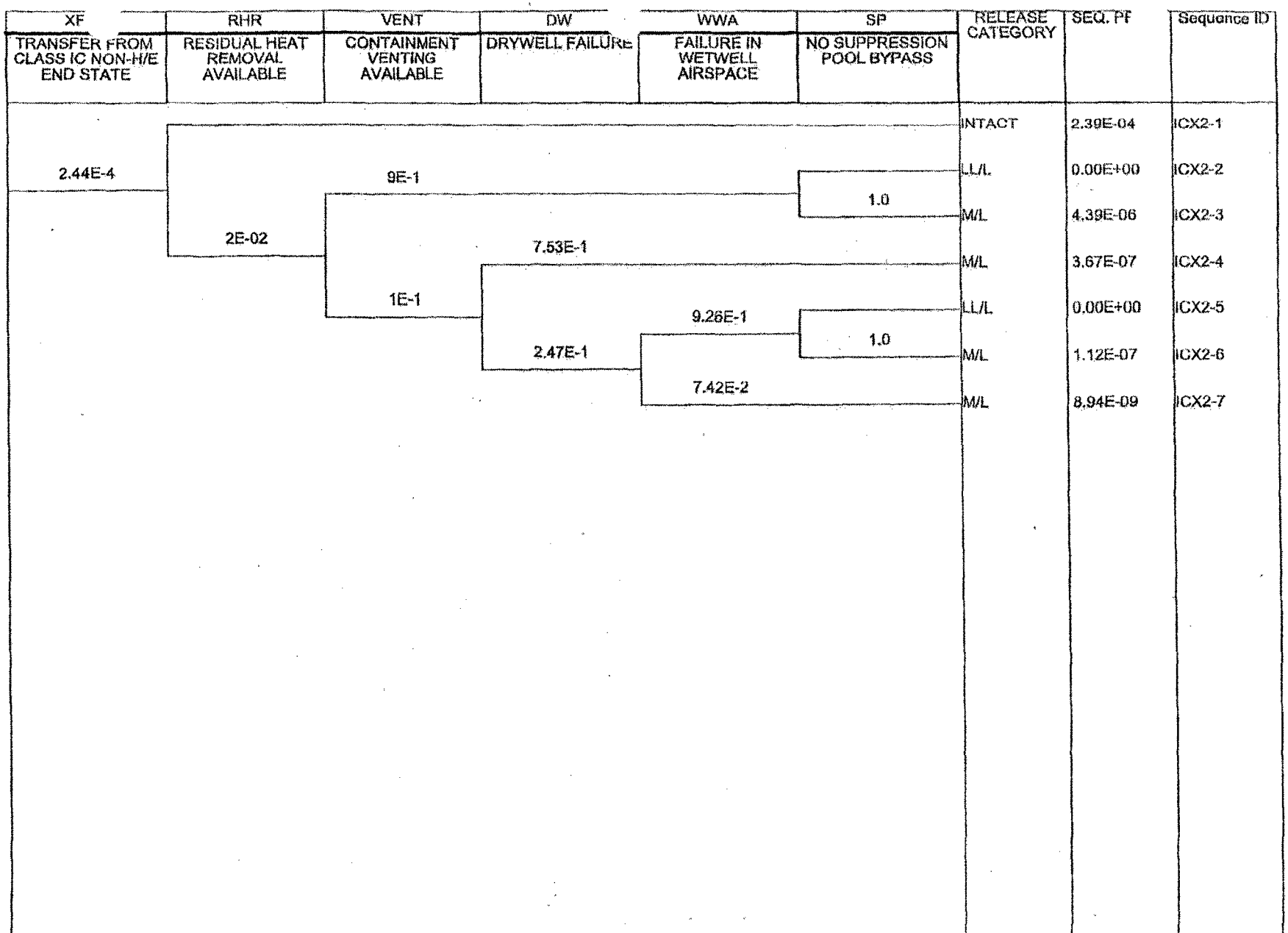

XFI
TRANSFER FROM CLASS IBL NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
2.77E-1
1E-03
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-3
2.1E-3
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
2.77E-01
2.49E-04
5.24E-07
2.09E-05
6.32E-06
1.33E-08
5.08E-07
IBLX1-1
IBLX1-2
IBLX1-3
IBLX1-4
IBLX1-5
IBLX1-6
IBLX1-7

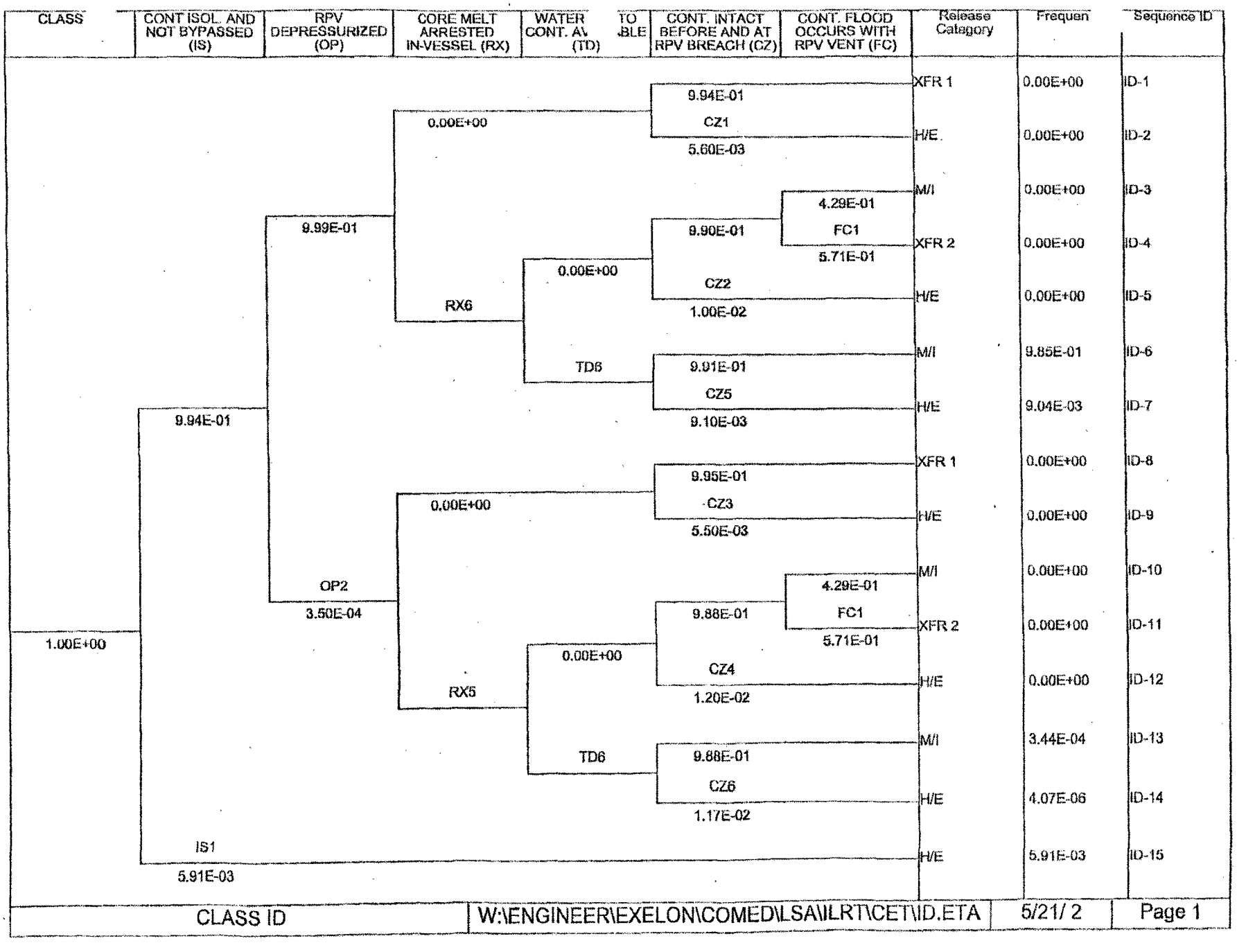

XF
TRANSFER FROM CLASS IBL NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
2.79E-1
1E-03
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
1.0
1.0
INTACT 2.79E-01 IBLX2-1
LL/L 0.00E+00 IBLX2-2
M/L 2.51E-04 IBLX2-3
M/L 2.10E-05 IBLX2-4
LL/L 0.00E+00 IBLX2-5
M/L 6.38E-06 IBLX2-6
M/L 5.11E-07 IBLX2-7



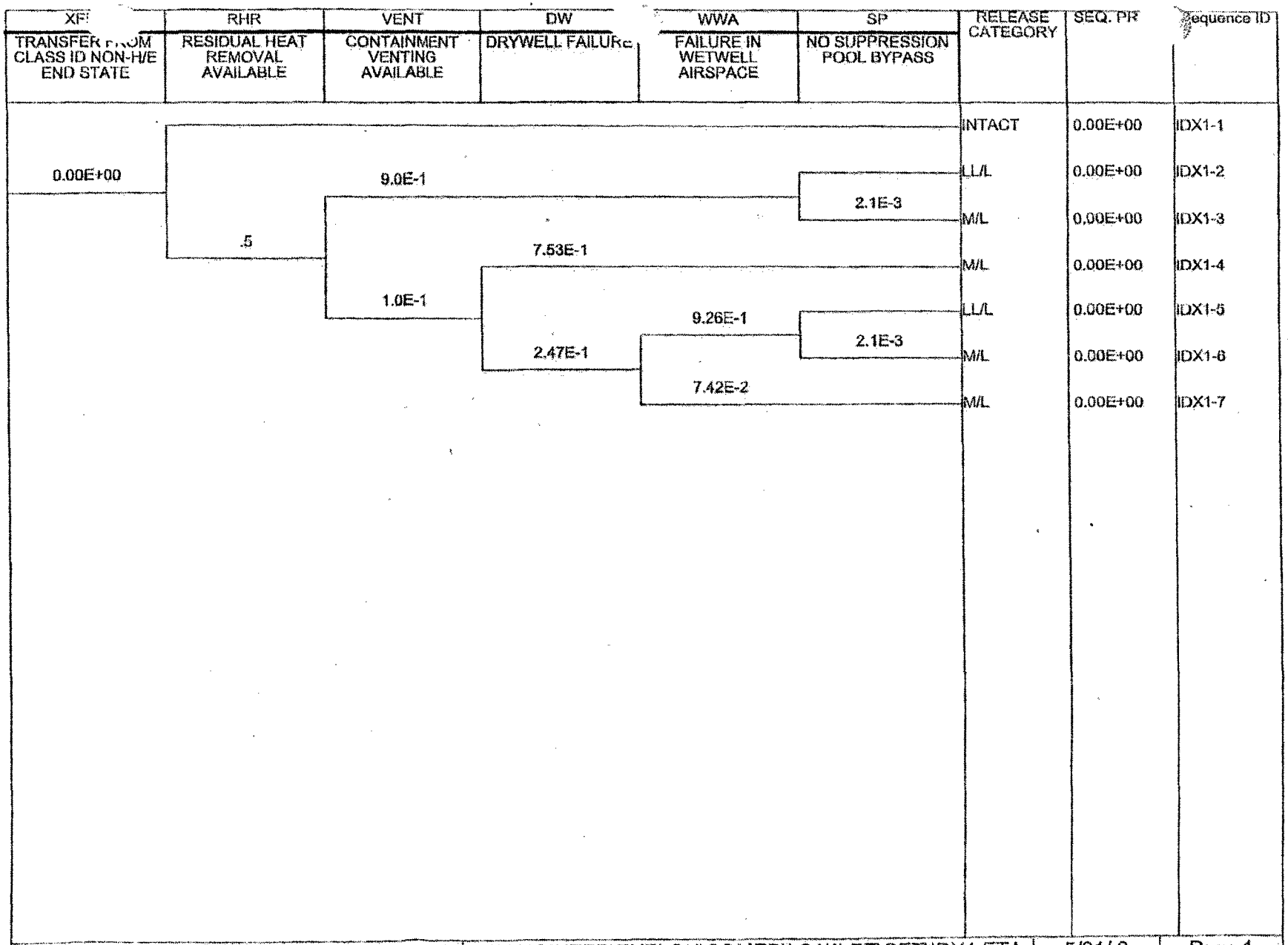
CLASS IC	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IC.ETA	5/21/ 2	Page 1

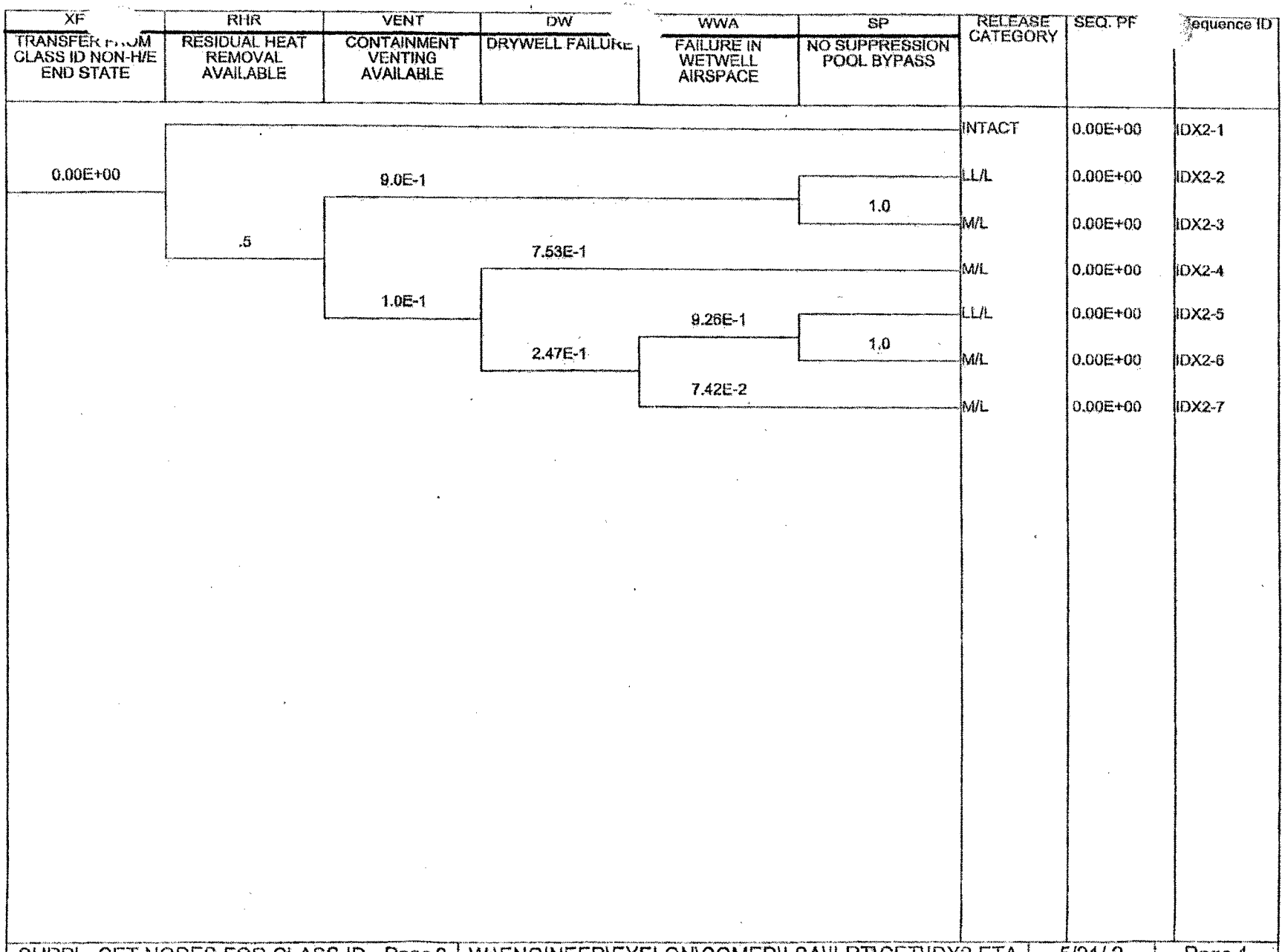

XF
TRANSFER FROM CLASS IC NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
9.88E-1
2E-02
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-3
2.1E-3
INTACT 9.68E-01 ICX1-1
LL/L 1.77E-02 ICX1-2
M/L 3.73E-05 ICX1-3
M/L 1.49E-03 ICX1-4
LL/L 4.51E-04 ICX1-5
M/L 9.49E-07 ICX1-6
M/L 3.62E-05 ICX1-7

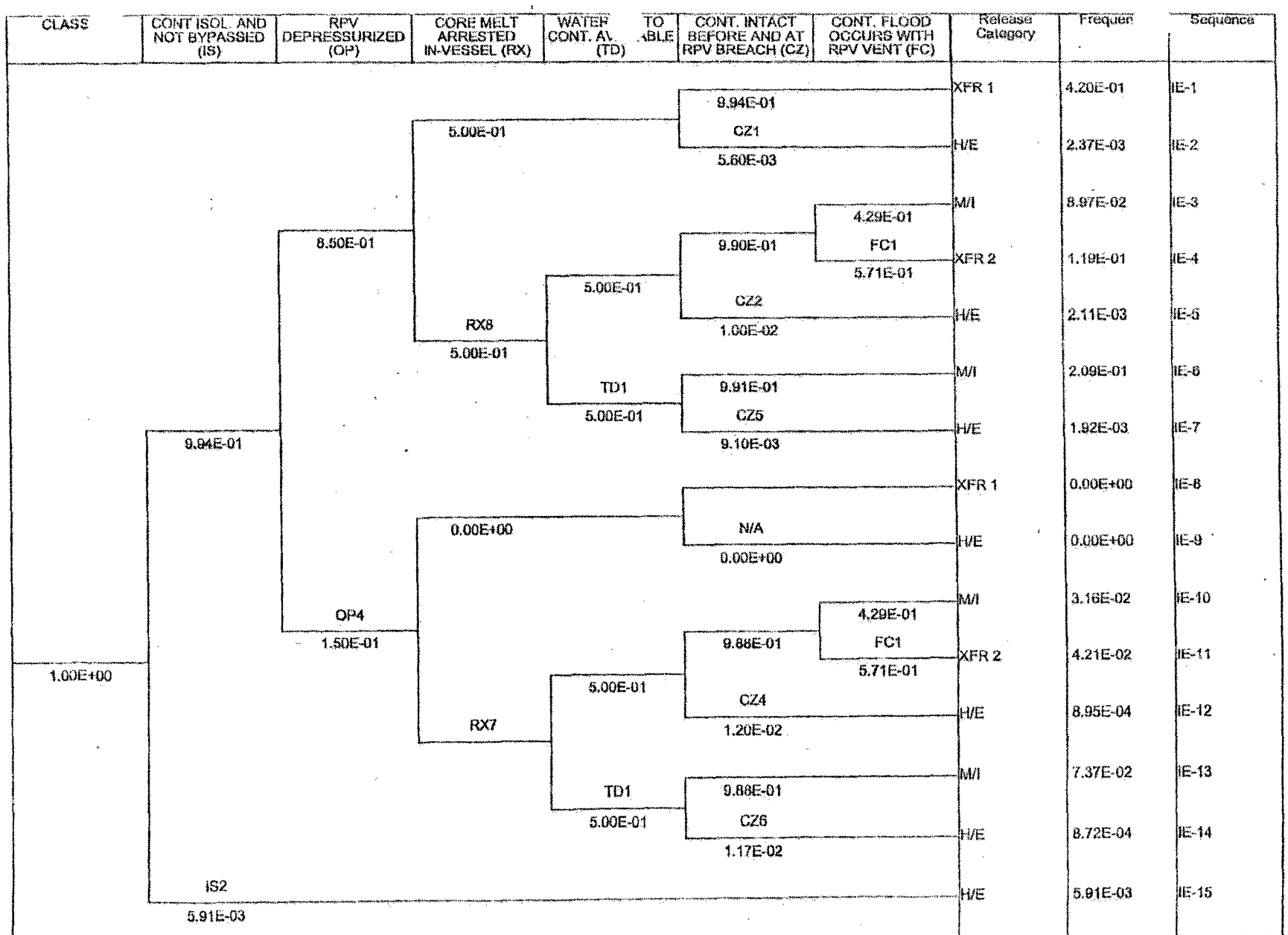

XF
TRANSFER FROM CLASS IC NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
2.44E-4
2E-02
9E-1
1.0
1E-1
7.53E-1
9.26E-1
1.0
2.47E-1
7.42E-2
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
2.39E-04
0.00E+00
4.39E-06
3.67E-07
0.00E+00
1.12E-07
8.94E-09
ICX2-1
ICX2-2
ICX2-3
ICX2-4
ICX2-5
ICX2-6
ICX2-7



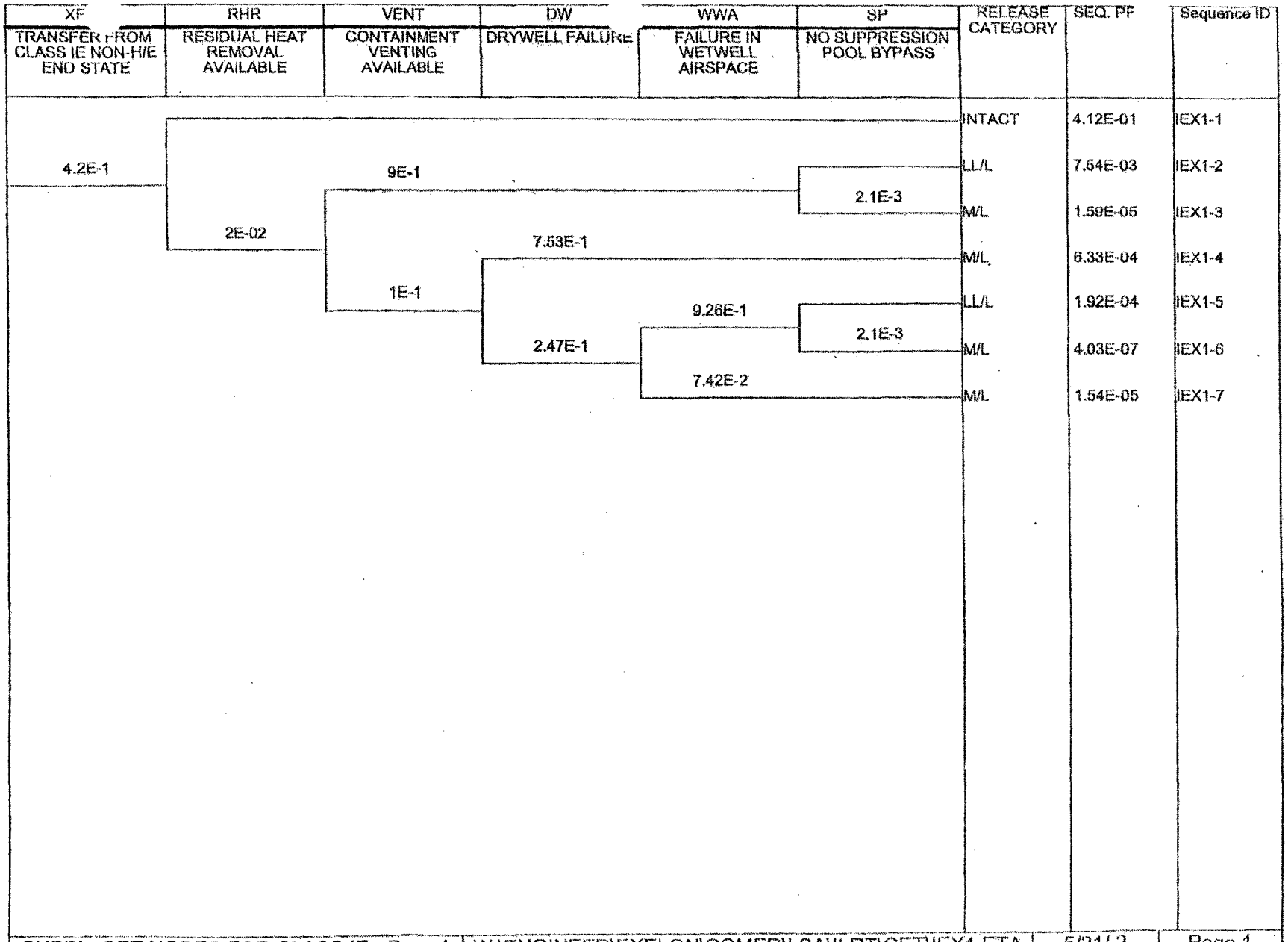
CLASS ID | W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\ID.ETA | 5/21/ 2 | Page 1

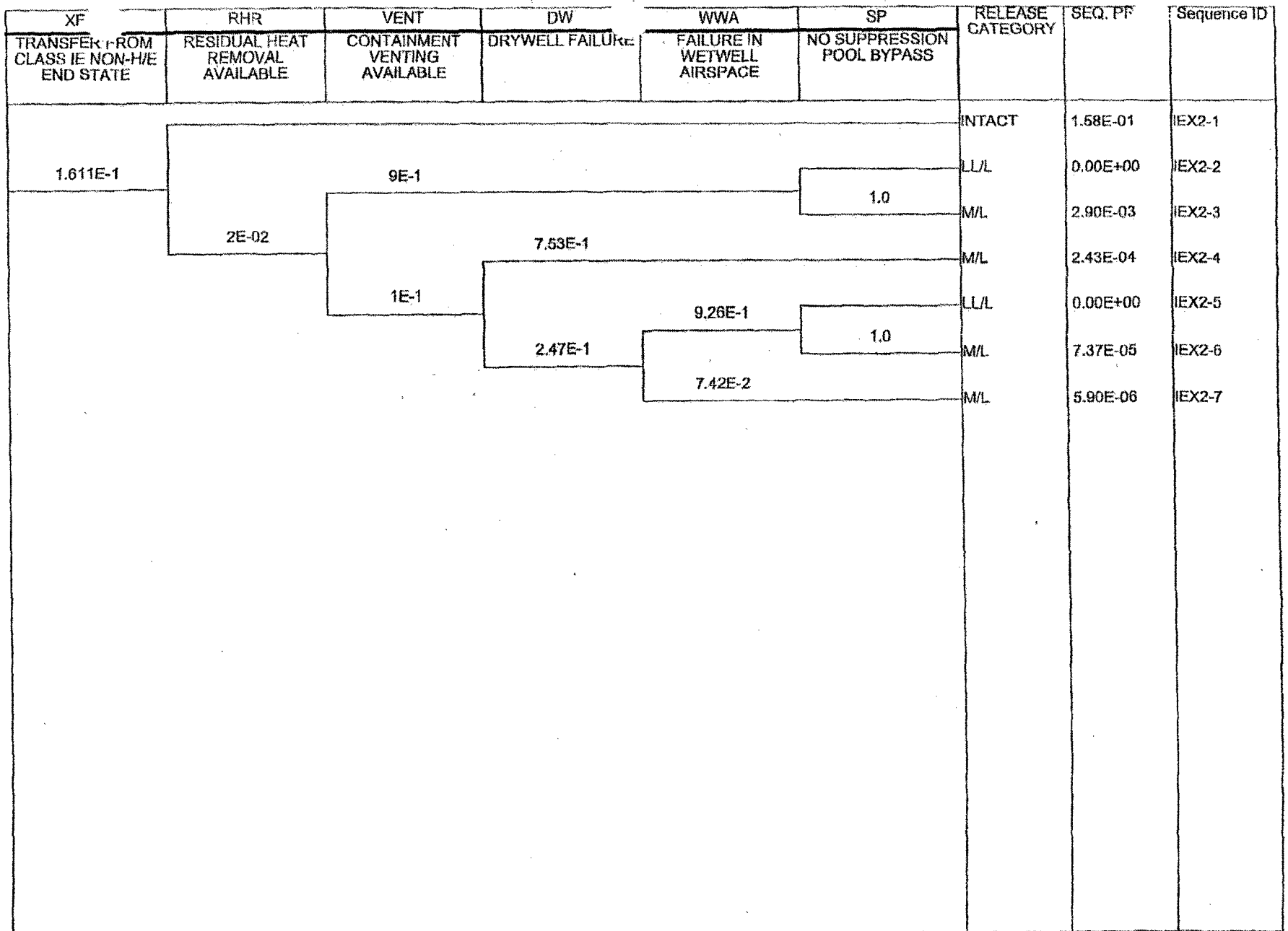

XF
TRANSFER FROM CLASS ID NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
0.00E+00
.5
9.0E-1
1.0E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-3
2.1E-3
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
IDX1-1
IDX1-2
IDX1-3
IDX1-4
IDX1-5
IDX1-6
IDX1-7

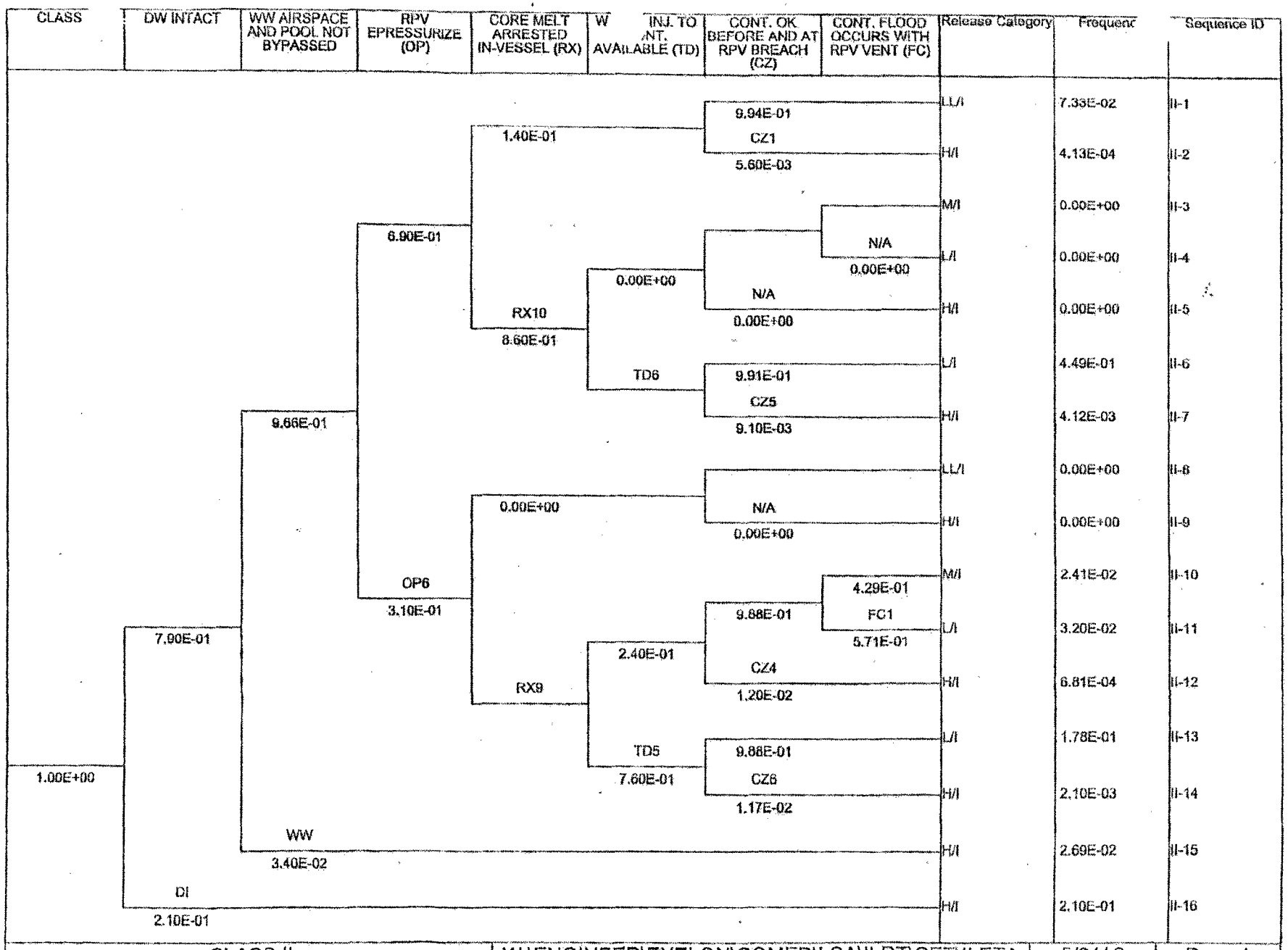

XF
TRANSFER FROM CLASS ID NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
0.00E+00
.5
9.0E-1
1.0E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
1.0
1.0
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
IDX2-1
IDX2-2
IDX2-3
IDX2-4
IDX2-5
IDX2-6
IDX2-7



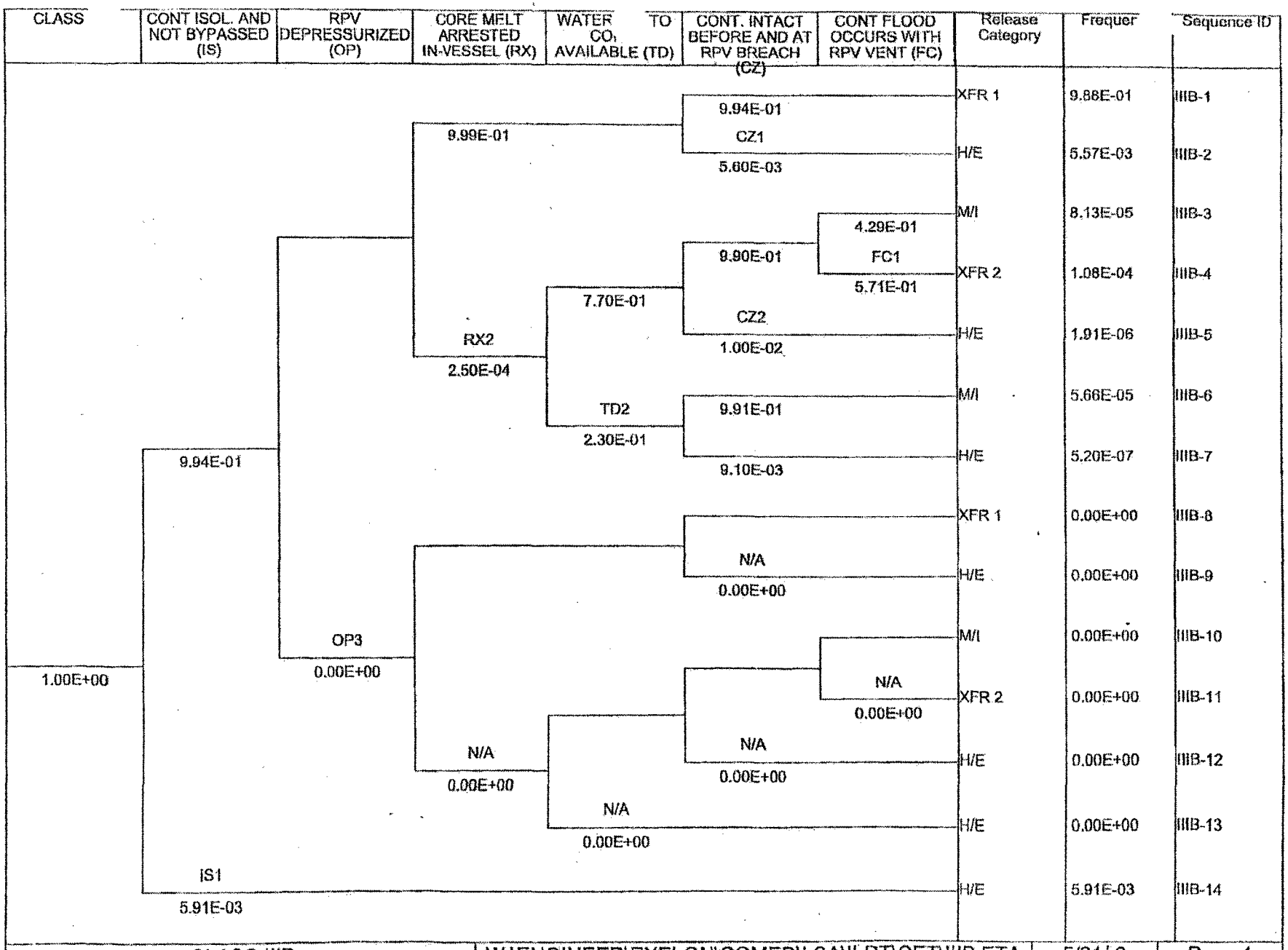
CLASS 1E	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IE.ETA	5/21/ 2	Page 1

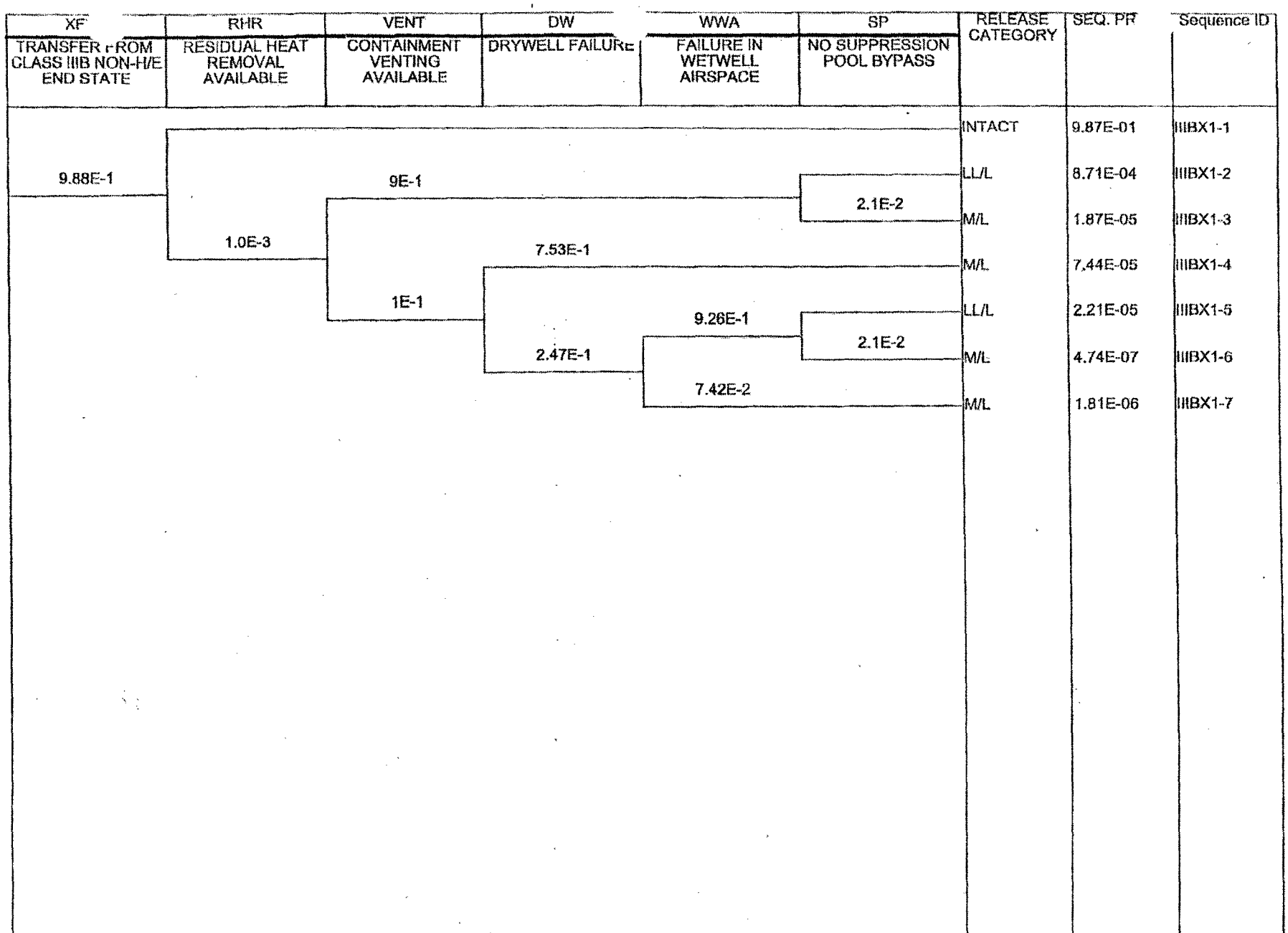

XF
TRANSFER FROM CLASS IE NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
4.2E-1
2E-02
9E-1
1E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-3
2.1E-3
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
4.12E-01
7.54E-03
1.59E-05
6.33E-04
1.92E-04
4.03E-07
1.54E-05
IEX1-1
IEX1-2
IEX1-3
IEX1-4
IEX1-5
IEX1-6
IEX1-7

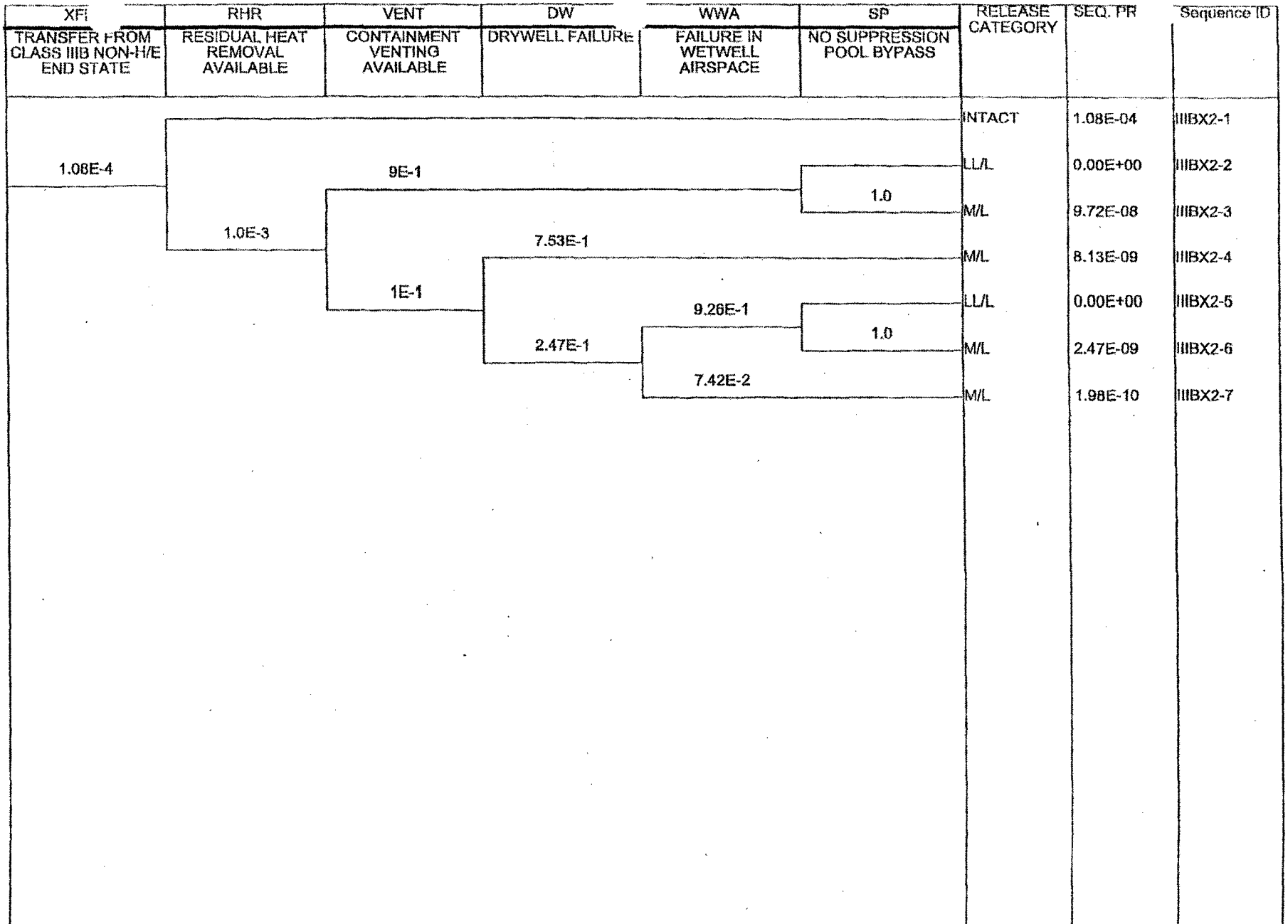

XF
TRANSFER FROM CLASS IE NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
1.611E-1
2E-02
9E-1
1E-1
1.0
7.53E-1
2.47E-1
9.26E-1
7.42E-2
1.0
INTACT 1.58E-01 IEX2-1
LL/L 0.00E+00 IEX2-2
M/L 2.90E-03 IEX2-3
M/L 2.43E-04 IEX2-4
LL/L 0.00E+00 IEX2-5
M/L 7.37E-05 IEX2-6
M/L 5.90E-06 IEX2-7

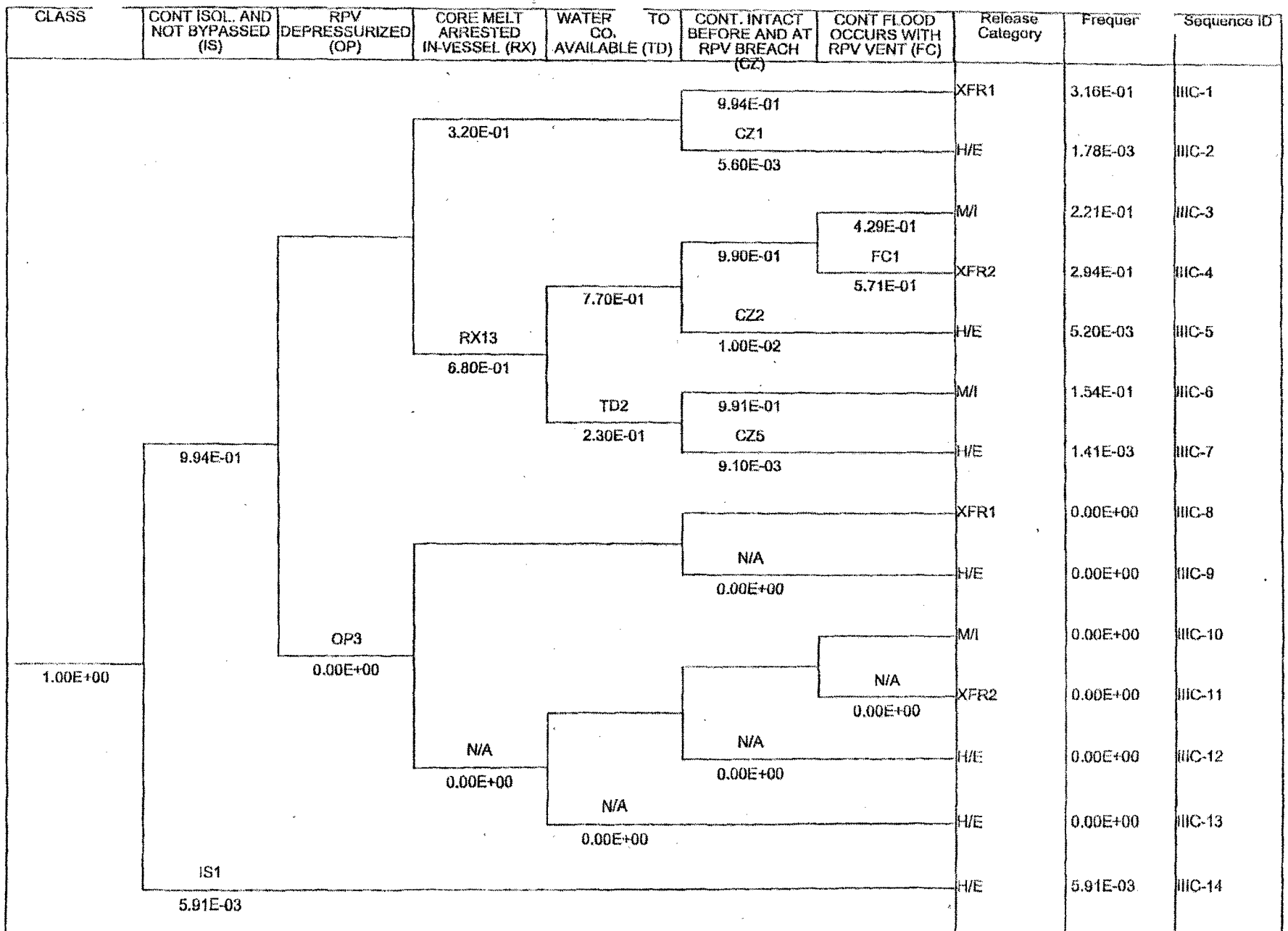


CLASS II	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\II.ETA	5/21/ 2	Page 1

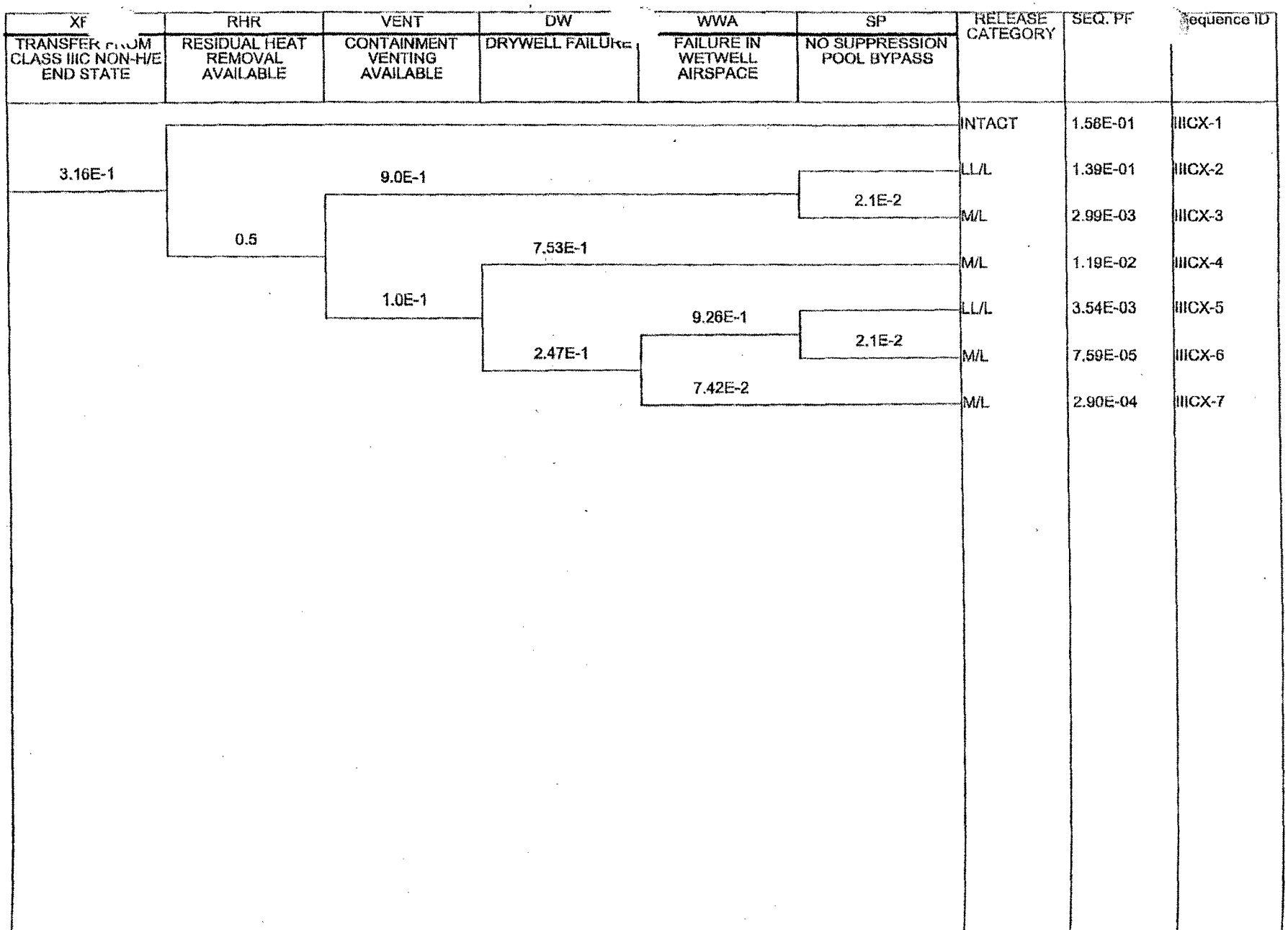

CLASS
CONT ISOL. AND NOT BYPASSED (IS)
RPV DEPRESSURIZED (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER TO CO AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequer
Sequence ID
1.00E+00
9.94E-01
IS1
5.91E-03
OP3
0.00E+00
9.99E-01
RX2
2.50E-04
N/A
0.00E+00
7.70E-01
TD2
2.30E-01
N/A
0.00E+00
9.94E-01
CZ1
5.60E-03
9.90E-01
CZ2
1.00E-02
9.91E-01
9.10E-03
N/A
0.00E+00
4.29E-01
FC1
5.71E-01
N/A
0.00E+00
N/A
0.00E+00
XFR 1 9.88E-01 IIIB-1
H/E 5.57E-03 IIIB-2
M/I 8.13E-05 IIIB-3
XFR 2 1.08E-04 IIIB-4
H/E 1.91E-06 IIIB-5
M/I 5.66E-05 IIIB-6
H/E 5.20E-07 IIIB-7
XFR 1 0.00E+00 IIIB-8
H/E 0.00E+00 IIIB-9
M/I 0.00E+00 IIIB-10
XFR 2 0.00E+00 IIIB-11
H/E 0.00E+00 IIIB-12
H/E 0.00E+00 IIIB-13
H/E 5.91E-03 IIIB-14

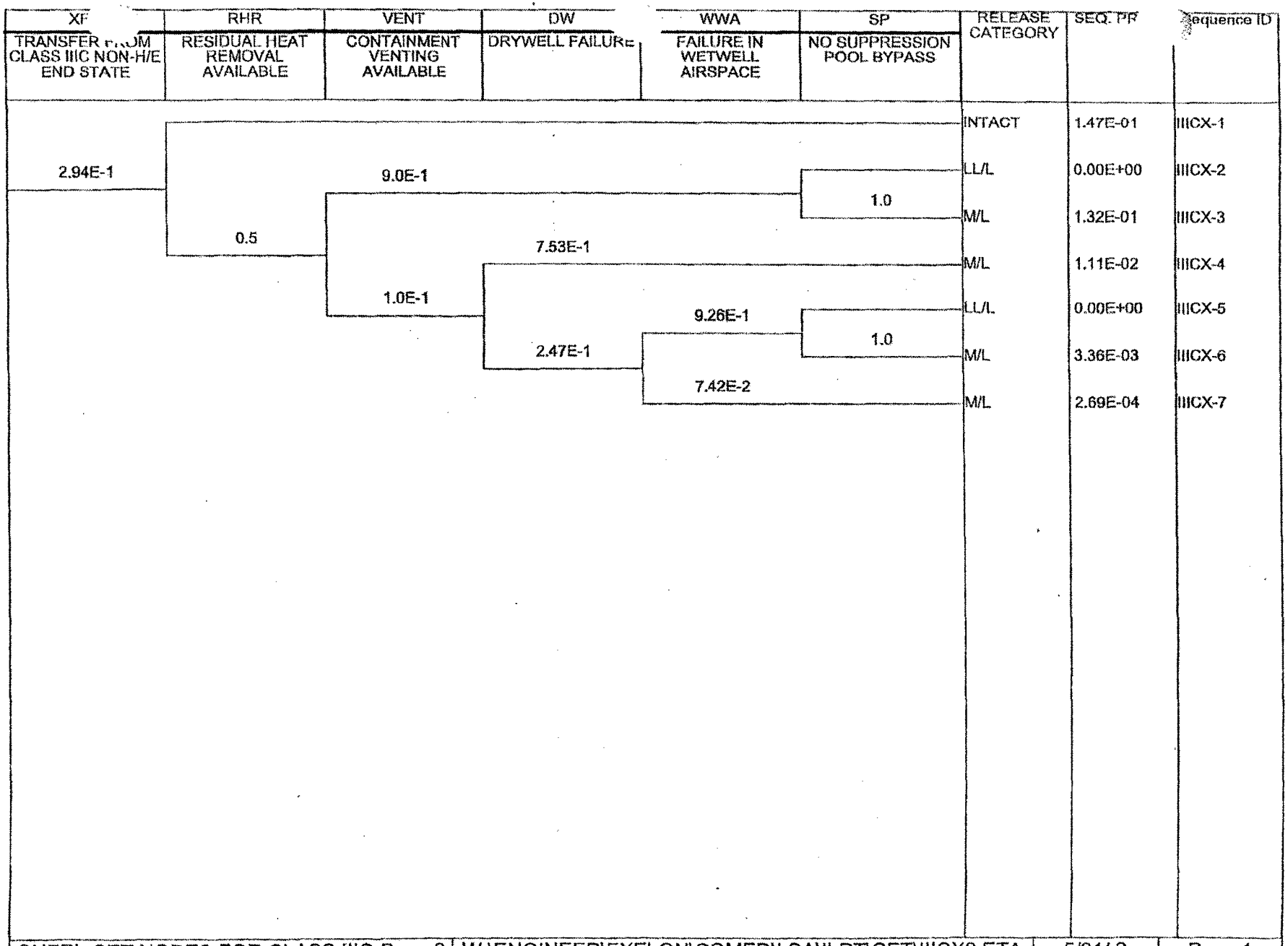

XF
TRANSFER FROM CLASS IIIB NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
9.88E-1
1.0E-3
9E-1
1E-1
2.1E-2
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-2
INTACT 9.87E-01 IIIBX1-1
LL/L 8.71E-04 IIIBX1-2
M/L 1.87E-05 IIIBX1-3
M/L 7.44E-05 IIIBX1-4
LL/L 2.21E-05 IIIBX1-5
M/L 4.74E-07 IIIBX1-6
M/L 1.81E-06 IIIBX1-7

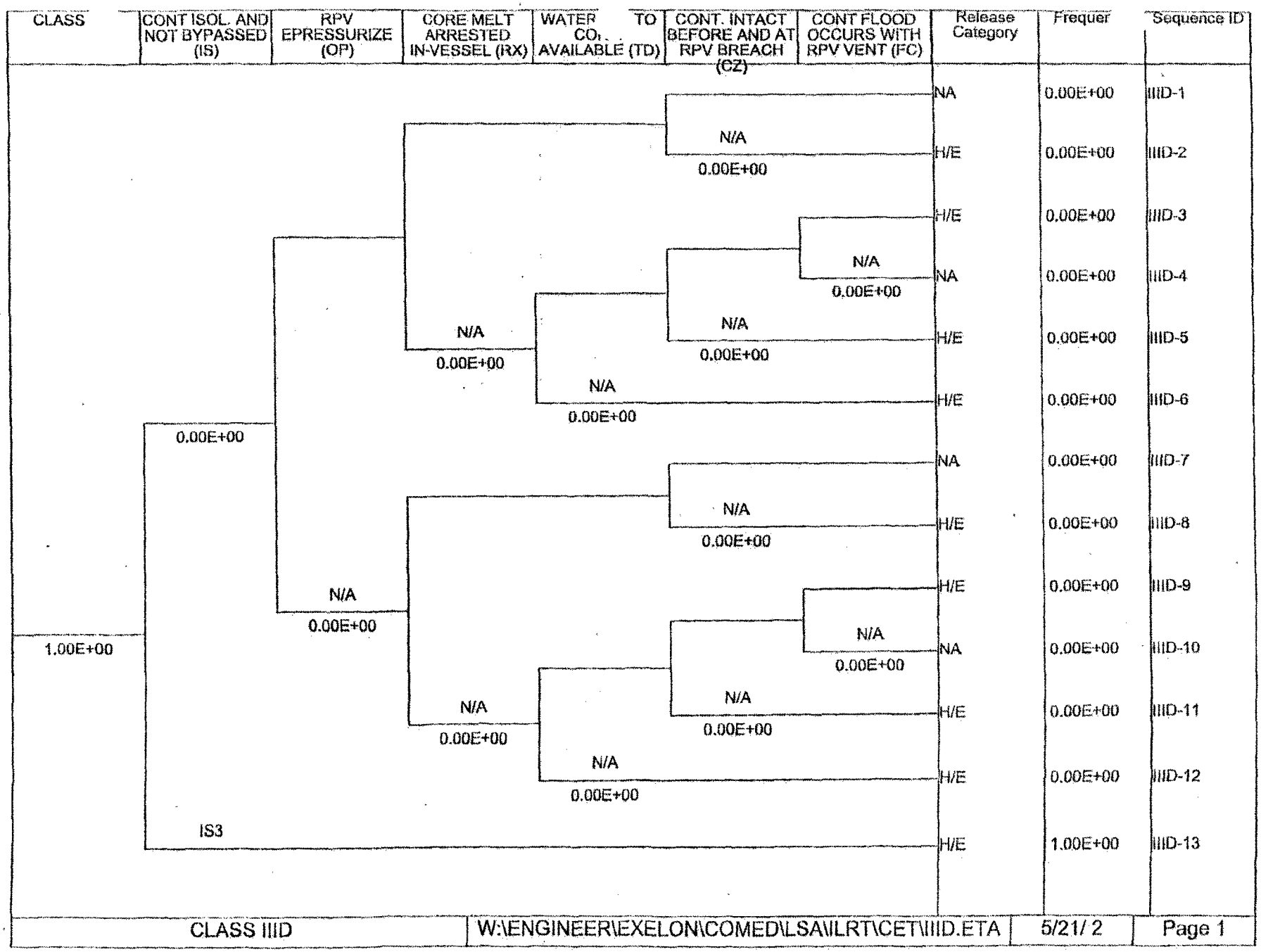

XFI
TRANSFER FROM CLASS IIIB NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
1.08E-4
1.0E-3
9E-1
1.0
7.53E-1
1E-1
9.26E-1
1.0
2.47E-1
7.42E-2
INTACT 1.08E-04 IIIBX2-1
LL/L 0.00E+00 IIIBX2-2
M/L 9.72E-08 IIIBX2-3
M/L 8.13E-09 IIIBX2-4
LL/L 0.00E+00 IIIBX2-5
M/L 2.47E-09 IIIBX2-6
M/L 1.98E-10 IIIBX2-7

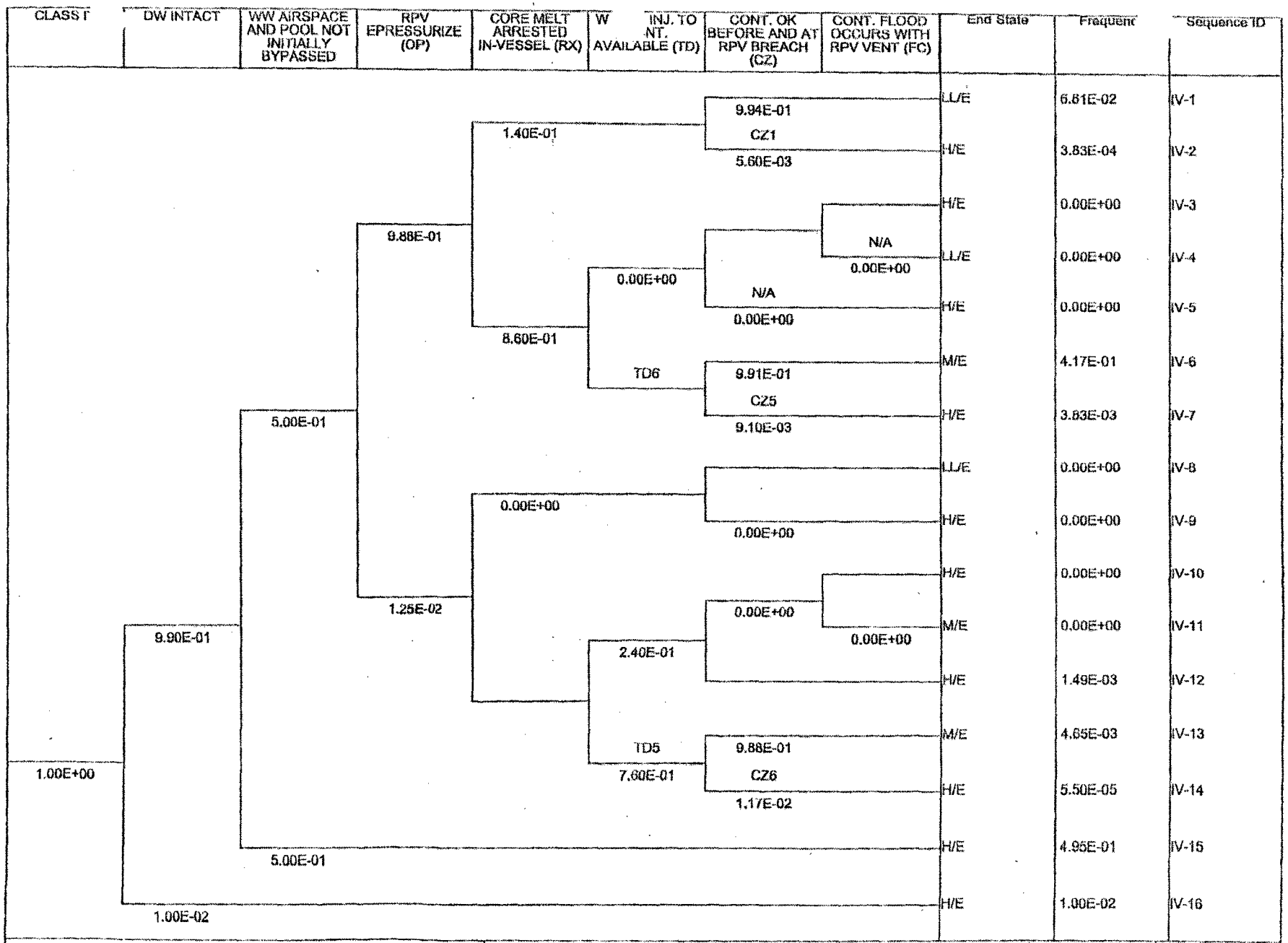


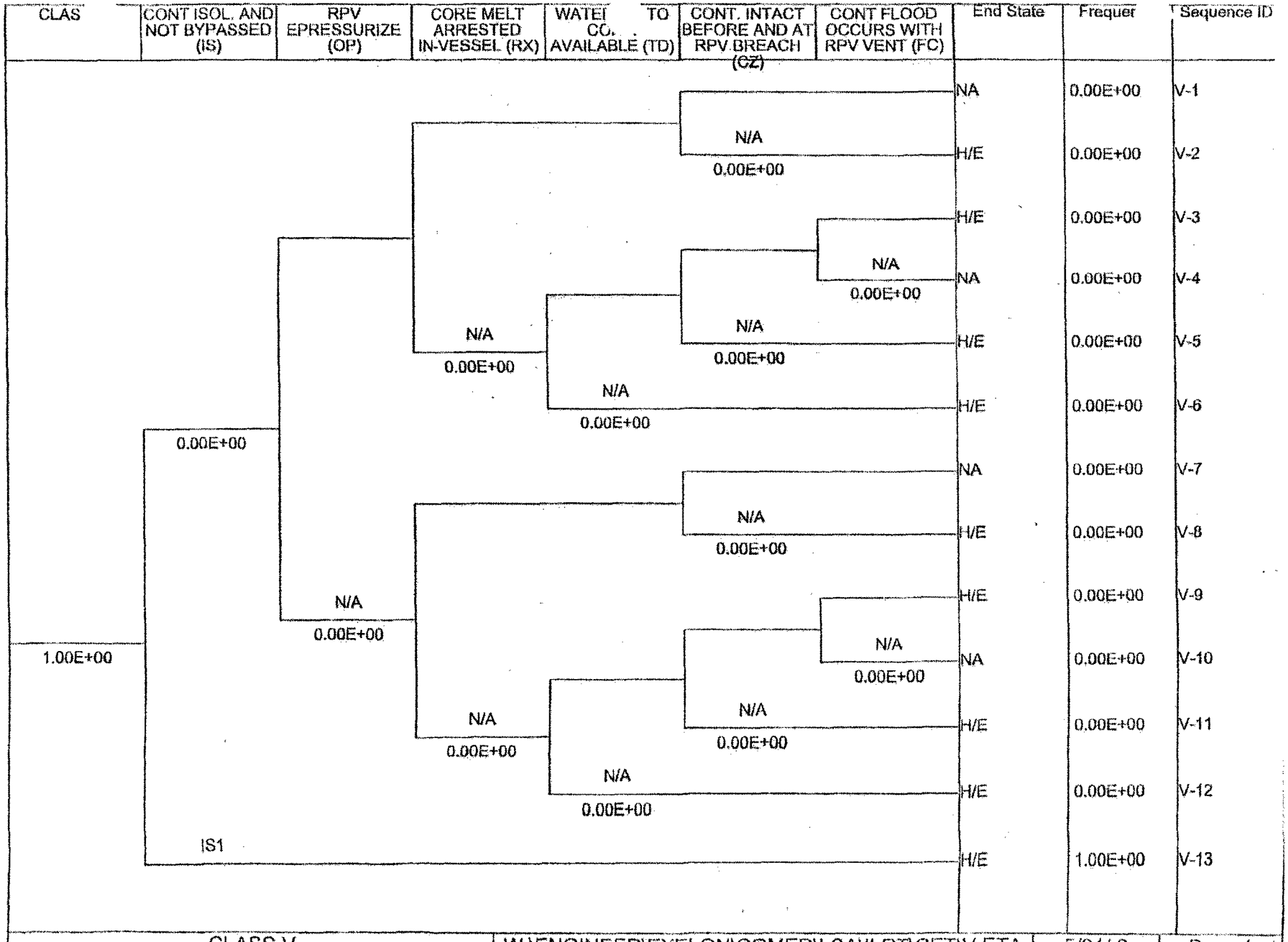
CLASS IIIC	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IIIC.ETA	5/21/ 2	Page 1


XF
TRANSFER FROM CLASS IIIC NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PF
Sequence ID
3.16E-1
0.5
9.0E-1
1.0E-1
7.53E-1
2.47E-1
9.26E-1
7.42E-2
2.1E-2
2.1E-2
INTACT
LL/L
M/L
M/L
LL/L
M/L
M/L
1.58E-01
1.39E-01
2.99E-03
1.19E-02
3.54E-03
7.59E-05
2.90E-04
IIICX-1
IIICX-2
IIICX-3
IIICX-4
IIICX-5
IIICX-6
IIICX-7


XF
TRANSFER FROM CLASS IIIC NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PR
Sequence ID
2.94E-1
0.5
9.0E-1
1.0
1.0E-1
7.53E-1
2.47E-1
9.26E-1
1.0
7.42E-2
INTACT 1.47E-01 IIICX-1
LL/L 0.00E+00 IIICX-2
M/L 1.32E-01 IIICX-3
M/L 1.11E-02 IIICX-4
LL/L 0.00E+00 IIICX-5
M/L 3.36E-03 IIICX-6
M/L 2.69E-04 IIICX-7


CLASS
CONT ISOL. AND NOT BYPASSED (IS)
RPV EPRESSURIZE (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER TO CO AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequer
Sequence ID
1.00E+00
0.00E+00
IS3
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
N/A
0.00E+00
NA
H/E
H/E
NA
H/E
H/E
NA
H/E
H/E
NA
H/E
H/E
H/E
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
1.00E+00
IIID-1
IIID-2
IIID-3
IIID-4
IIID-5
IIID-6
IIID-7
IIID-8
IIID-9
IIID-10
IIID-11
IIID-12
IIID-13


CLASS I
DW INTACT
WW AIRSPACE AND POOL NOT INITIALLY BYPASSED
RPV EPRESSURIZE (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
W INJ. TO NT. AVAILABLE (TD)
CONT. OK BEFORE AND AT RPV BREACH (CZ)
CONT. FLOOD OCCURS WITH RPV VENT (FC)
End State
Frequenc
Sequence ID
1.00E+00
9.90E-01
1.00E-02
5.00E-01
5.00E-01
9.88E-01
1.25E-02
1.40E-01
8.60E-01
0.00E+00
0.00E+00
2.40E-01
7.60E-01
TD6
TD5
9.94E-01
CZ1
5.60E-03
N/A
0.00E+00
N/A
0.00E+00
9.91E-01
CZ5
9.10E-03
0.00E+00
0.00E+00
0.00E+00
0.00E+00
9.88E-01
CZ6
1.17E-02
LL/E 6.81E-02 IV-1
H/E 3.83E-04 IV-2
H/E 0.00E+00 IV-3
LL/E 0.00E+00 IV-4
H/E 0.00E+00 IV-5
M/E 4.17E-01 IV-6
H/E 3.83E-03 IV-7
LL/E 0.00E+00 IV-8
H/E 0.00E+00 IV-9
H/E 0.00E+00 IV-10
M/E 0.00E+00 IV-11
H/E 1.49E-03 IV-12
M/E 4.65E-03 IV-13
H/E 5.50E-05 IV-14
H/E 4.95E-01 IV-15
H/E 1.00E-02 IV-16


CLAS
CONT ISOL. AND NOT BYPASSED (IS)
RPV EPRESSURIZE (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATEI TO CO. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT FLOOD OCCURS WITH RPV VENT (FC)
End State
Frequer
Sequence ID
1.00E+00
0.00E+00
IS1
N/A
0.00E+00
NA
H/E
V-1
V-2
V-3
V-4
V-5
V-6
V-7
V-8
V-9
V-10
V-11
V-12
V-13
1.00E+00